42

FFN #01-010

ARKANSAS NUCLEAR ONE		Page 1
FORM TITLE: 10CFR50.59 DETERMINATION	**FORM NO.** 1000.131A	**REV.** 003-04-0

This Document contains 2 Pages.

Document No. TAP 01-1-001 Rev./Change No. 0

Title Temporary Cooling Water to E-28C ICW Cooler

Brief description of proposed change: **Provide Temporary Cooling Water to E-28C while SW return header is OOS.**

Will the proposed Activity:

1. Require a change to the Operating License including:
   - Technical Specifications (excluding the bases)? Yes☐ No☒
   - Operating License? Yes☐ No☒
   - Confirmatory Orders? Yes☐ No☒
2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:
   - SAR (multi-volume set for each unit)? Yes☒ No☐
   - Core Operating Limits Report? Yes☐ No☒
   - Fire Hazards Analysis? Yes☐ No☒
   - Bases of the Technical Specifications? Yes☐ No☒
   - Technical Requirements Manual? Yes☐ No☒
   - NRC Safety Evaluation Reports? Yes☐ No☒
3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒
4. Result in a potential impact to the environment? (Complete Environmental Impact Determination of this form.) Yes☐ No☒
5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒
6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒
7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7?
   - QAPM? Yes☐ No☒
   - E-Plan? Yes☐ No☒
8. Does this review depend on future NRC approval of other actions? (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. TAP 01-1-001 Rev./Change No. 0

Basis for Determination (Questions 1, 2, & 3): See Continuation Page.

☐ Proposed change does not require 10CFR50.59 Evaluation per Attachment 1, Item #____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in questions 1, 2 and 3. If search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS: **50.59 Unit 1**	**ALL (intermediate cooling, icw, spent fuel w/15 boil*, service w/15 spent)**
MANUAL SECTIONS: **U1 SAR**	**9.3.1; 9.3.2; 9.4; 9.4.2.6; 9.6.1.3; 14; Table 9.8**
FIGURES: **U1 SAR**	**Fig. 9-6, Fig. 9-7, Fig. 9-8, Fig. 9-9, Fig, 9-16**

James H. Crabill — Certified Reviewer's Signature

James Crabill — Printed Name

1/29/01 — Date

Reviewer's certification expiration date: 4/21/01

Assistance provided by:

Printed Name	Scope of Assistance	Date

Search Scope Review Acceptability (NA, if performed by Technical Reviewer per 1000.006)

T Grant Ehren — Certified Reviewer's Signature

T GRANT EHREN — Printed Name

2/8/01 — Date

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

Document No. TAP 01-1-001 Rev./Change No. 0

Complete the following Determination. If the answer to any item below is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

Document No. TAP 01-1-001 Rev./Change No. 0

10CFR50.59 Review Continuation Page

This temporary alteration provides cooling water as the ultimate heat sink for spent fuel pool cooling during 1R16 while the reactor is defueled and the Service Water Return header is out of service for maintenance work. Service Water will be used as the cooling source to ICW cooler, E-28C and a temporary return header will allow the SW to return to the lake via the Auxiliary Cooling Water return header. ICW will, using normal system configuration and alignment, be used for cooling of the Unit 1 Spent Fuel Pool. Calculations documented in ER 003327E101 indicate this arrangement will provide adequate cooling to maintain the bulk Spent Fuel Pool temperature at or below 150 deg. F during the maximum expected pool heat load.

1. This temporary alteration affects the acheivable Service Water system flow to the ICW Cooler (2500 gpm vs 2800 gpm normal) due to the temporary return line configuration. These issues are beyond the scope of the Units' operating license documents. Therefore, this temporary alteration will not require a change to the Operating License documents.

2. This temporary alteration will result in information contained in the Unit 1 SAR, more specifically the system description and system configuration for cooling water return from ICW cooler E-28C, being no longer true or accurate. A 50.59 safety evaluation will be performed.

3. This temporary alteration does not involve a test or experiment. Therefore, this change does not involve a test or experiment not described in the SAR.

4. This temporary alteration will not result in a potential impact to the environment. See page 3 of 4 of this determination.

5. This temporary alteration does not involve the processing of radioactive material outside of the Aux. Bldg, Reactor Bldg, or low level Radwaste storage bldg, nor does it create a new pathway outside of the monitored ventilation or drainage pathways.

6. This temporary alteration does not involve any potential impact to equipment, facilities, or anything else associated with dry fuel storage. Dry fuel activities will not be in progress during the outage.

7. This temporary alteration has no impact on the QAMO or the E-Plan.

8. This temporary alteration does not depend on future NRC approval of other actions.

FORM TITLE: 10CFR50.59 SAFETY EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 1 Page.

Document No. TAP 01-1-001 Rev./Change No. 0 10CFR50.59 Eval. No. FFN#01-010 (Assigned by PSC)

Title **Temporary Cooling Water to E-28C ICW Cooler**

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   This activity is not an accident initiator nor does it affect any SAR evaluated accidents. Coolant loss due to boiling, in the unlikely event of a complete loss of cooling, would not be significant as a backup system for supplying water to the pool is provided through a temporary connection to the Seismic Category 1 service water system, as identified in the SAR. The Unit 1 service water system will remain available to makeup to the pool if required. Sufficient time to establish this connection exists as pool boiling would not occur for a minimum of 4.1 hours given an initial pool temperature of 150 deg. F as documented in ER 003327E101. Seismic Class 1 makeup to the spent fuel pools remains available from the service water system. Consequently, this activity will not increase the probability of an accident previously evaluated in the SAR.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   This temporary alteration has no impact to the radiation dose consequences of any accident evaluated in the SAR. As a result, this temporary change will not increase the consequences of an accident previously evaluated in the SAR.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   This activity does not impact any important-to-safety equipment. As such, implementation of this temporary change will not increase the probability of a malfunction of equipment important to safety.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   This temporary alteration does not affect equipment important to safety. As a result, installation of this temporary alteration will not result in equipment important to safety failing in a manner which increases the dose consequences of an accident nor will this activity have a bearing on the radiological release consequences of an accident assuming a malfunction of equipment important to safety. Therefore, installation of this temporary alteration will not increase the consequences of a malfunction of equipment important to safety.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   The credible failure modes of this temporary alteration are failure of a hose resulting in flooding and loss of cooling to ICW resulting in a loss of cooling to the Spent Fuel Pool. Both of these scenarios are bounded by the SAR and have contingency actions available to mitigate the

consequences of either. Thus, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created by the implementation of this activity.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   This activity does not impact equipment important to safety. As such, this temporary alteration will not affect the type of malfunctions of equipment important to safety and thereby, will not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the SAR.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

   There are no margins of safety in the Technical Specification bases that are affected by this activity. Therefore, installation of this temporary alteration will not result in the reduction of any margin of safety as defined in the bases of the technical specifications.

Certified Reviewer's Signature	Printed Name	Date
James Crabill	James Crabill	1/29/01

Reviewer's certification expiration date: 4/21/01

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: _____ Date: 2/14/01

43

FFN# 01-012

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This document contains 4 pages

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA

Title **Surface Area of Contact Between VCC-018 and the Storage Pad Less Than Required by VSC-24 SAR (use-as-is)**

Brief description of proposed change:

While placing ventilated fuel storage cask 18 on the Dry Fuel Storage pad, the surface area of contact between the VCC and the concrete pad was measured as less than the required 20.97 sq. ft (reference calculations in the VSC SAR paragraph 3.4.4.2.1). The actual (measured) area of contact is approximately 5 sq. ft. The condition is to be reviewed herein for acceptability as a candidate for "use-as-is".

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☐ No☒

   Core Operating Limits Report? Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☒ No☐

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7?

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions? (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. **VSC SAR Docket No. 72-1007** Rev./Change No. **0AA**

Basis for Determination (Questions 1, 2, & 3):

See continuation page.

☐ Proposed change does not require 10CFR50.59 Evaluation per Attachment 1, Item #____. (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in questions 1, 2 and 3. If search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.

Document	Section
LRS:	
50.59 Common Unit	**All (searched for: VSC, VCC, dry fuel)**
ANO-1 SAR 1998 Revision	**All (searched for: VSC, VCC, dry fuel)**
ANO-2 SAR 1998 Revision	**All (searched for: VSC, VCC, dry fuel)**
MANUAL SECTIONS:	
Unit 1 SAR	**SAR 9.6.1.1**
	9.3.1
	6.3.1
Unit 2 SAR	**SAR 9.1.2A**
	15.1.23.2.2K
	12.1.2.8
	Table 15.1.0-1
	Table 15.1.23-2
FIGURES:	
None	

Darrell Williams — Certified Reviewer's Signature
Darrell R. Williams — Printed Name
2-8-01 — Date

Reviewer's certification expiration date: 12-29-2001

Assistance provided by:

Printed Name	Scope of Assistance	Date
Kirk L. Dixon	**Research and preparation**	**February 1, 2001**

Search Scope Review Acceptability (NA, if performed by Technical Reviewer per 1000.006)

Certified Reviewer's Signature — Jay Gray (Hellwood) — Printed Name — 2-8-2001 — Date

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA

10CFR50.59 Review Continuation Page

Question 1:

The condition does not require a change to the Operating License including Technical Specifications, Operating License, or Confirmatory Orders since the specific VSC-24 requirements are not listed in these documents. The VSC SAR, the VSC SER, and the VSC Certificate of Conformance list the specific requirements and will be addressed in the associated 10CFR72.48 determination and evaluation.

Question 2:

The condition does not result in information in the (station) SAR, Core Operating Limits Report, Fire Hazards Analysis, Bases of the Technical Specifications, Technical Requirements Manual, or NRC (station) Safety Evaluation Reports (including drawings and text) being no longer true or accurate. The condition does not violate a requirement stated in any of these documents. These statements are true, since the specific VSC-24 requirements are not addressed in these documents. The VSC SAR, the VSC SER, and the VSC Certificate of Conformance list the specific requirements and will be addressed in the associated 10CFR72.48 determination and evaluation.

Question 3:

The condition does not involve a test or experiment not described in the SAR. The requirements for the condition are addressed in the VSC-24 SAR, not the station SAR. Therefore, if any experiment or test is affected, then it would need to be addressed in the VSC-24 SAR.

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA

Complete the following Determination. If the answer to any item below is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA 10CFR72.48 Eval. No. FFN401-012 (Assigned by PSC)

Title Surface Area of Contact Between VCC-018 and Storage Pad Less Than Required by VSC-24 SAR (use-as-is)

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 TO PROCEDURE 1000.131 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1.	Will the probability of an accident previously evaluated in the cask SAR be increased?	Yes ☐	No ☒
2.	Will the consequences of an accident previously evaluated in the cask SAR be increased?	Yes ☐	No ☒
3.	Will the probability of a malfunction of equipment important to safety be increased?	Yes ☐	No ☒
4.	Will the consequences of a malfunction of equipment important to safety be increased?	Yes ☐	No ☒
5.	Will the possibility of an accident of a different type than any previously evaluated in the cask SAR be created?	Yes ☐	No ☒
6.	Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the cask SAR be created?	Yes ☐	No ☒
7.	Will the margin of safety as defined in the bases for the cask Conditions for System Use be reduced?	Yes ☐	No ☒

Darrell Williams 2-8-01

Certified 10CFR72.48 Reviewer's Signature | Printed Name: Darrell Williams | Date: 2-8-01

Reviewer's certification expiration date: 12-29-2001

Assistance provided by:

Printed Name	Scope of Assistance	Date
Kirk L. Dixon	Research and preparation	February 1, 2001

PSC review by: _____ Date: 2/15/01

FORM TITLE:	FORM NO.	REV.
10CFR72.48 EVALUATION	1000.132B	000-00-0

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA

Title Surface Area of Contact Between VCC-018 and Storage Pad Less Than Required by VSC-24 SAR (use-as-is)

10CFR72.48 Review Continuation Page

Basis for Responses:

Question 1:
The proposed activity will not increase the probability of an accident previously evaluated in the cask SAR. The accidents evaluated in the VSC SAR encompass fuel pin failure, maximum heat load, MSB drop, tornado, flood, earthquake, accidental pressurization, fresh fuel loading, and full blockage of air vents. The probability of occurrence of these accidents is not affected in any manner by the decreased surface area of contact under the VSC. Except for the potential blockage of air vents, these accident causes are natural phenomena or conditions controlled by procedure and are not a function of the VCC contact area condition. The probability for the blockage of air vents is not increased since the probability of VCC failure is not increased.

Question 2:
The proposed activity will not increase the consequences of an accident previously evaluated in the cask SAR. The consequences of the evaluated accidents relate to radiological dose and heat transfer. The reduced surface area does not increase the contained waste material nor decrease the shielding of the cask system. Therefore, the condition does not increase the consequences of an accident previously evaluated.

Question 3:
The proposed activity will not increase the probability of a malfunction of equipment important to safety. The cask has been shown to be stable in a slightly tilted condition and therefore the probability of tip over is not increased. Other equipment important to safety is not affected in any fashion by the reduced surface contact.

Question 4:
The proposed activity will not increase the consequences of a malfunction of equipment important to safety. The only equipment affected by the surface contact area is the VCC and its ability to support the weight of the loaded MSB. The consequences of a VCC malfunction relate to loss of radiological shielding and heat transfer. The reduced surface area does not increase the contained waste material nor decrease the shielding of the cask system. The reduced surface area does not increase the heat transfer requirements to keep the fuel cool. Therefore, the condition does not increase the consequences of a malfunction of equipment important to safety.

Question 5:
The proposed activity will not create the possibility of an accident of a different type than any previously evaluated in the cask SAR. The evaluated condition does not change the system content, form or function and does not significantly change the system fit. The condition does not change the way the cask is handled and does not change the interface with radioactive materials or station equipment important to safety. Therefore, the condition does not create the possibility of an accident of a different type than previously evaluated.

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA

Title Surface Area of Contact Between VCC-018 and Storage Pad Less Than Required by VSC-24 SAR (use-as-is)

10CFR72.48 Review Continuation Page

Basis for Responses:

Question 6:
The proposed activity will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the cask SAR. The condition changed the potential tilt of the VSC assembly but is still encompassed within the design tip over calculations with respect to weight and center of gravity. The strength of the cask is not compromised. No other equipment important to safety will interface with this cask while resting on the storage pad such that no new possibility for equipment malfunction can be created.

Question 7:
The proposed activity will not reduce the margin of safety as defined in the bases for the Conditions for Cask Use. The cask, if used as is, will meet all design criteria stipulated in the Conditions for Cask Use. The only difference for this one-time use-as-is condition is that the load carrying area of the VCC bottom is reduced to below the desired area shown in VSC SAR section 3.4.4.2.1, but the area is still above the calculated minimum shown in VSC SAR section 3.4.3.1.

Conclusion:

There is not an unreviewed safety question as a result of the low surface area of contact between VSC-018 and the storage pad surface. It is acceptable for this VSC to be "used-as-is".

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA This document contains 7 pages

Title Surface Area of Contact Between VCC-018 and Storage Pad Less Than Required by VSC-24 SAR (use-as-is)

Brief description of proposed change:
While placing ventilated fuel storage cask 18 on the Dry Fuel Storage pad, the surface area of contact between the VCC and the concrete pad was measured as less than the required 20.97 sq. ft (reference calculations in the VSC SAR paragraph 3.4.4.2.1). The actual (measured) area of contact is approximately 5 sq. ft. The condition is to be reviewed herein for acceptability as a candidate for "use-as-is".

Will the proposed Activity:

1.	Require a change to the cask		
	Certificate of Compliance?	Yes ☐	No ☒
	Conditions for System Use (including Bases)?	Yes ☐	No ☒
2.	Result in a significant increase in occupational exposure related to cask use?	Yes ☐	No ☒
3.	Result in a significant unreviewed environmental impact? (List and attach 10CFR50.59 Determination containing Environmental Impact Determination.)	Yes ☐	No ☒
	10CFR50.59 Review Title: Surface Area of Contact Between VCC-018 and Storage Pad Less Than Required by VSC-24 SAR (use-as-is) Date: 2/8/2001		
4.	Result in information in the cask SAR (including text, tables, figures, and drawings) or SER being either		
	(a) No longer true or accurate, or		
	(b) violate a requirement stated in the document?	Yes ☒	No ☐
5.	Involve a test or experiment not described in the cask SAR?	Yes ☐	No ☒

Basis for Determination:

See Continuation Page attached.

☐ The proposed change does not require 10CFR72.48 Evaluation per Attachment 1, Item #_____, of Procedure 1000.131 (if checked, note appropriate item # and send LDCR to Licensing).

FORM TITLE:	FORM NO.	REV.
10CFR72.48 DETERMINATION	**1000.132A**	**000-00-0**

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA

Title Surface Area of Contact Between VCC-018 and Storage Pad Less Than Required by VSC-24 SAR (use-as-is)

Search Scope:

List sections reviewed in the Cask Licensing Basis Documents specified in questions 1, 4 and 5. If search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures, tables, or drawings). Attach and distribute a completed LDCR if Cask LBD changes are required.

Document	Section
LRS:	
VSC C of C	**All (searched for: VCC w/10 bottom, VSC w/10 bottom)**
VSC SAR	**All (searched for: VCC w/10 bottom, VSC w/10 bottom)**
VSC SER	**All (searched for: VCC w/10 bottom, VSC w/10 bottom)**
MANUAL SECTIONS:	
VSC C of C	**All (Table of Contents)**
	Section 1.2.11
VSC SAR	**Section 3.4.3.1**
	Section 3.4.4.2.1

FIGURES:
None

Darrell Williams — Darrell Williams — 2-5-01
Certified Reviewer's Signature — Printed Name — Date

Reviewer's certification expiration date: 12-29-2001

Assistance provided by:

Printed Name	Scope of Assistance	Date
Kirk L. Dixon	**Research and preparation**	**February 1, 2001**

Search Scope Review Acceptability (NA, if performed by Technical Reviewer per 1000.006)

[signature] — Jay Gray Hellwood — 2-8-2001
Certified Reviewer's Signature — Printed Name — Date

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA

Title Surface Area of Contact Between VCC-018 and Storage Pad Less Than Required by VSC-24 SAR (use-as-is)

10CFR72.48 Review Continuation Page

Basis for Determination:

The strength of the VCC bottom is adequate to support the weight of the fuel , MSB, and the VCC even with a reduced surface area of contact. Section 3.4.3.1 of the VSC SAR calculates the maximum load capability of the VCC bottom, considering the maximum shear stress and ~~bending~~ bearing stress in the concrete without strength credit for the steel plate on the bottom surface. It assumes a lift from the bottom by eight jacks, each one being six inches in diameter. The resulting allowable loads of 633 kips (~~bending~~ bearing) and 528 kips (shear) are well above the actual load of 279 kips, with the shear load being the limiting condition. It is clear that any bearing surface area greater than the 226 square inches presented by the eight jacks (six inches diameter each) would be sufficient to properly support the loaded VCC. Therefore, the surface area contact condition present with VSC-018 (approximately 700 square inches) is acceptable with regard to the VCC strength. KAD 2-15-01

For the storage pad concrete, if the reduced area of contact is very conservatively assumed to adversely affect the capacity of the pad to support the VCC, then punch through could occur at one or more of the three points of contact. If one or two points punch through, the tilt of the cask is safe from tip over since the cask is designed for a tilt of much greater than 3/4 inch. If all three points of contact punch through, then the area of contact is increased to the design area, since the VCC bottom would then be in nearly full contact. The cask would be in a safe condition.

Therefore, it is acceptable to use the cask VCC-018 in its current condition, with reduced surface area contact with the storage pad. "Use-as-is" is acceptable.

Question 1:
The proposed activity will not require a change to the cask Certificate of Compliance (C of C) or the Conditions for Cask Use (CCU's). The C of C and the CCU's do not list the minimum surface area of contact requirement specifically since this requirement is below the level of detail therein. Note that the phrase "Conditions for System Use" was changed administratively to "Conditions for Cask Use" in Revision 1 of the C of C.

Question 2:
The proposed activity will not result in a significant increase in occupational exposure related to cask use. All radiological barriers remain unaffected and fully functional. There is no danger of cask tip over since the fully loaded cask geometry is not adversely affected by this slight change in gap between the cask bottom and the storage pad surface. The gap needed to adversely affect the tip over calculation is much greater than the 3/4 inch present with this condition. Passive radiological shielding and barriers are not changed in any fashion. The bottom portion of the VCC concrete has no streaming considerations with regard to this gap.

Question 3:
The proposed activity will not result in a significant unreviewed environmental impact of any kind. The condition does not change the release, removal, emission, concentration, or placement of any materials beyond those already evaluated for dry fuel storage. The "use-as-is" condition is identical to the design condition with respect to the environmental impact.

Therefore, no NRC approval is required for the "use-as-is" conclusion of this evaluation.

Document No. VSC SAR Docket No. 72-1007 Rev./Change No. 0AA

Title: Surface Area of Contact Between VCC-018 and Storage Pad Less Than Required by VSC-24 SAR (use-as-is)

10CFR72.48 Review Continuation Page

Basis for Determination:

Question 4:
The proposed activity will result in information in the cask SAR being either no longer true or accurate or will violate a requirement stated in the SAR. The paragraph 3.4.4.2.1 ("VCC Dead Load") addresses the stress in the VCC concrete bottom due to the dead load of the MSB and fuel being taken through the concrete bottom over the surface area of the MSB. This conservative means to calculate the bottom stress is acceptable and simple if the contact area is as normal, that is, in excess of 20.97 sq. ft. (the area of an MSB footprint). However, the condition for VCC-018 is not the normal contact area in excess of 20.97 sq. ft, but rather a fraction thereof. Justification for "use-as-is" is addressed in the 10CFR72.48 evaluation. The cask SER is not affected since the condition is below the level of detail provided therein.

Question 5:
The proposed activity does not involve a test or experiment not described in the cask SAR. This activity does not involve a test or experiment of any kind and all applicable testing previously required remain unchanged.

Note that since Question 4 is affirmative, an evaluation in accordance with 10CFR72.48 is required for this "use-as-is" conclusion.

44

ER002875N101	ARKANSAS NUCLEAR ONE		Page 1
FORM TITLE: 10CFR50.59 DETERMINATION		FORM NO. 1000.131A	REV. 003-04-0

PAGE 6 REV. 0 This Document contains 3 Pages.

Document No. ER002875N101 Rev./Change No. 0

Title Reactor Building Temporary Monitoring Removal

Brief description of proposed change:

Locate, identify and remove approximately 52 cables routed from temporary temperature and flow detectors to the Westronics multiplexer located at elevation 376' in Reactor Building installed per DCP-87-1095 and Special Work Plan WP 1409.45.

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report? Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7?

   QAPM? Yes☐ No☒
   No☒
   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions? (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. ER002875N101 Rev./Change No. 0

Basis for Determination (Questions 1, 2, & 3):

The original purpose of the installation was to evaluate the effectiveness of the corrective actions implemented to reduce RB temperatures. Subsequently, this temporary monitoring function is no longer required and can be removed since alternate monitoring has been provided and RB temperatures have been reduced satisfactorily. This modification will not require a change to Unit 1 Operating License but does result in information on SAR figures listed below to be revised to indicate temporary monitoring detectors as spared in place. An LDCR has been submitted to licensing for the proposed changes. Finally, this removal is not a test nor an experiment.

☐ Proposed change does not require 10CFR50.59 Evaluation per Attachment 1, Item # ____. (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in questions 1, 2 and 3. If search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS: Unit 1 50.59	ALL (Temporary Monitoring, JCA, Reactor Building Temperature Monitoring, Monitoring Temperature, temperatures, rb temperature, temporary monitoring)

O 863 1-31-01

MANUAL SECTIONS: N/A

FIGURES: 4-1, 5-7, 6-3, 7-20, 6-13

Douglas A. Bruce Douglas A. Bruce D.A. BRUCE 1/16/01
Certified Reviewer's Signature Printed Name Date

Reviewer's certification expiration date: 2/25/01

Assistance provided by:

Printed Name	Scope of Assistance	Date

Search Scope Review Acceptability (NA, if performed by Technical Reviewer per 1000.006)

Lee A. Pinkett Lee A. Pinkett 1/31/01
Certified Reviewer's Signature Printed Name Date

ENVIRONMENTAL IMPACT DETERMINATION
(UNIT 1 and UNIT 2)

Document No. ER002875N101 Rev./Change No. 0

Complete the following Determination. If the answer to any item below is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to non radiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

FORM TITLE: 10CFR50.59 SAFETY EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 2 Pages.

Document No. ER002875N101 Rev./Change No. 0 10CFR50.59 Eval. No. FFN#01-013 (Assigned by PSC)

Title Reactor Building Temporary Monitoring and Cable Removal

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

Abstract:
The original purpose of the installation of Reactor Building (RB) temporary monitoring was to evaluate the effectiveness of the corrective actions implemented to reduce RB temperatures. Subsequently, this temporary monitoring function is no longer required and can be removed since alternate monitoring is provided.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   The accident analysis for Unit 1 SAR does not consider temporary monitoring in the Reactor Building. The accident initiators evalued in the SAR accident analyses are not affected by temporary monitoring used for tracking improvements to RB cooling. This modification does not invalidate the failure modes outlined in the SAR, nor does this activity increase the frequency of any accident initiator. Therefore, the probability of an accident previously evaluated in the SAR is not increased.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   The removal of temporary RB monitoring does not affect the consequences of any accident previously evaluated accident. This ER does not invalidate any accident assumption nor consequences outlined in the SAR, since the temporary monitoring does not provide any control nor alarm function, nor does it interface with any system which would affect Offsite dose rates. Therefore, the consequences of an accident previously evaluated in the SAR are not increased by this modification.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   This modification does not alter nor affect the function or capability of any equipment required to perform a safety related function. The removal of the temporary monitoring does not affect the operation of any existing safety equipment. This modification has no impact on system reliability, separation, seismic features, specification nor safety loads. Therefore, the probability of a malfunction of equipment important to safety remains unchanged.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   This modification removes temporary monitoring in Unit 1 Reactor Building which is no longer required to be operational, having satisfied its initial temporary function. Therefore, the consequences of a malfunction of equipment important to safety are not increased.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   Removal of RB temporary monitoring cannot initiate any new type of accident previously evaluated in SAR. The temporary system has been previously evaluated and is bounded by existing accident analyses. Therefore, its removal cannot create the possibility of an accident of a different type than any previously evaluated in the SAR.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   The ability of any equipment to perform their safety related functions is not compromised by this modification. The removal of this temporary monitoring system creates no new equipment failures nor failure scenarios. Therefore, the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR is not created.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

   The Reactor Building temporary monitoring system nor its components are not discussed in the bases for any Tech Spec reviewed. Removing this system will not impact any Tech Spec bases, and therefore the margin of safety is not reduced.

Certified Reviewer's Signature	Douglas A. Bruce Printed Name	1-24-00 Date

Reviewer's certification expiration date: 2/25/01

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: _____ Date: 2/19/01

45

FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 5 Pages.

Document No. Procedure 1305.034 Rev./Change No. 000-00-0

Title **MAKEUP TANK RELIEF PATH ISOLATION CONTROLS**

Brief description of proposed change:

This procedure will allow the temporary isolation of the MU tank relief path in order to perform maintenance on the waste gas system. Compensatory measures will be taken consisting of specific directions to the Control Board Operators for controlling MUT level and pressure. Pressure/level in the makeup tank will be controlled using manual bleed or by aligning the makeup pumps to the BWST.

Will the proposed Activity:

1. Require a change to the Operating License including:
   - Technical Specifications (excluding the bases)? Yes☐ No☒
   - Operating License? Yes☐ No☒
   - Confirmatory Orders? Yes☐ No☒
2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:
   - SAR (multi-volume set for each unit)? Yes☒ No☐
   - Core Operating Limits Report? Yes☐ No☒
   - Fire Hazards Analysis? Yes☐ No☒
   - Bases of the Technical Specifications? Yes☐ No☒
   - Technical Requirements Manual? Yes☐ No☒
   - NRC Safety Evaluation Reports? Yes☐ No☒
3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒
4. Result in a potential impact to the environment? (Complete Environmental Impact Determination of this form.) Yes☐ No☒
5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒
6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒
7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7?
   - QAPM? Yes☐ No☒
   - E-Plan? Yes☐ No☒
8. Does this review depend on future NRC approval of other actions? (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. Procedure 1305.034 Rev./Change No. 000-00-0

Basis for Determination (Questions 1, 2, & 3):

This change is beyond the level of detail specified in the Operating License or any OL documents. This temporary condition deals with maintenance activity due to a degraded condition (waste gas system component) which will be corrected and then the affected system will be restored to its previous condition (as described in the SAR), therefore this temporary change will not make the SAR or any SAR documents permanently untrue or inaccurate. The Makeup Tank, although purchased to ASME section III, is not safety related and is maintained seismic category I only to protect the integrity of the isolation valve CV-1275. The isolation of the vent and relief path has been determined to be acceptable by Engineering, therefore this is not a test and providing an equivalent protection does not constitute an experiment not described in the SAR.

☐ Proposed change does not require 10CFR50.59 Evaluation per Attachment 1, Item #____. (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in questions 1, 2 and 3. If search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS:	
50.59 Unit 1	All (PSV-1249, CV-12*, CV-4614, GZ*, ABV*, hydrogen over*, makeup tank, makeup w/5 relief, tank w/5 relief, tank w/5 vent)
MANUAL SECTIONS:	
ANO-1 SAR	4.2.3-5-4.2.3.8, 6.1.2.4.6, 9.1, Table 9-1, Table 9-2, 11.1.3.6.2, 14.1.2.4.1, Appendix A A7.1
ANO-1 Proposed ITS	3.4.11
ANO-1 Proposed ITS Bases	3.4.11, 3.4.12, 3.4.13, 3.4.15
ANO-1 Tech Spec	3.1.6, 3.2, 4.0.5
ANO-1 TS Bases	3.1.4, 3.1.6, 3.2
ANO-1 NSE	chapter 9
FIGURES:	
ANO-1 SAR	Fig. 11-1, 9-3

John Richardson — Certified Reviewer's Signature

John Richardson — Printed Name

2/12/01 — Date

Reviewer's certification expiration date: 06/08/2002

Assistance provided by:

Printed Name	Scope of Assistance	Date
None		

Search Scope Review Acceptability (NA, if performed by Technical Reviewer per 1000.006)

Alan Cox — Certified Reviewer's Signature

Alan Cox — Printed Name

2/14/01 — Date

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

Document No. Procedure 1305.034 Rev./Change No. 000-00-0

Complete the following Determination. If the answer to any item below is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

Document No. Procedure 1305.034 Rev./Change No. 000-00-0

10CFR50.59 Review Continuation Page

This procedure controls and implements conditions and measures to be taken when the makeup tank relief and vent path are not available.

With the exception of isolating the makeup tank vent and relief path, all of the information presented in the procedure consisting of notes, cautions and operating instructions is taken from currently reviewed and approved procedure 1104.002 and note 7.1 is from 91-R-1018-02 Att. 1 pages 176 – 179 EOP Setpoint Basis Document. Therefore the notes and cautions and operating instructions will not impact any of the LBDs.

This procedure is intended to be used during temporary maintenance or surveillance activities and not for permanent changes to the plant. This condition will exist during maintenance and surveillance activities and does not represent a permanent change to the facility. The isolation and restoration of the vent path will be considered a temporary alteration to SSCs and temporary alteration controls will be implemented in this procedure. The impact on an attached SSC (the makeup tank) requires the performance of a 50.59 review per 1000.131 and this represents that review to determine the impact on the attached SSC.

While the makeup tank vent and relief path are isolated by the closure of ABV-40 or other methods such that T-4 is no longer capable of being connected to T-76, the SAR Figure 11-1 will be made temporarily untrue. Since this is a temporary activity associated with maintenance or surveillance and will not be a change to the design of the plant, a correction to the SAR is not warranted. Those sections of the SAR which detail that venting of the T-4 is an activity that raises the amount of waste gas collected or contributes to total radioactive gas generation over an operating cycle will also not require revision as this is not a permanent change to the facility.

An additional consideration would arise should the control room operator be unable to complete his tasks of providing makeup tank overpressure protection. The function of the makeup tank during those situations where the control room operator would not be present (such as control room fires, DBAs) is summarized in EAR 92-003:

"Makeup Tank T-4 is an ASME Section III component which is not needed for safe shutdown. In the event of a rupture of the tank, the resulting release would be significantly lower than the calculated release for the gas decay tank. 10CFR100 limits and the more stringent NUREG-0800 limits would not be approached."

This procedure retains the instruction to isolate the T-4 at an 18 inch level. Since the procedure will require isolation of T-4 on low level, the inability of the Makeup Tank Vent Valve (CV-1257) to vent will not impact emergency operations.

This change does not impact environmental controls such that an environmental impact evaluation would be required. Since a control room operator will be used in lieu of PSV-1249 and an approved procedure will be utilized during this activity which does not allow any change in Radwaste processing a radiological safety evaluation is not required. The details of the emergency plan and QAMO/QAPM will not require evaluation under 10CFR50.54 since this activity will be performed consistent with their details. The change does not impact the VSC or any VSC facilities, therefore a 10CFR72.48 review is not required.

Document No. Procedure 1305.034 Rev./Change No. 000-00-0 10CFR50.59 Eval. No. FFN#01-015 (Assigned by PSC)

Title **MAKEUP TANK RELIEF PATH ISOLATION CONTROLS**

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   The makeup tank and relief path isolation may impact the following SAR accidents:

   Moderator Dilution Accident (14.1.2.4): The nominal moderator dilution event considered is the pumping of water with zero boron concentration from the makeup tank to the RCS. Isolation of the makeup tank vent and relief path will have no impact on the SAR detailed controls concerning prevention of moderator dilution accident. Establishment of the control room operator as an equivalent measure of the operation of the makeup tank relief valve will not result in more frequent dilutions or other additions to the makeup tank. The control room operator will take action in the event of makeup tank overpressure and all compensatory actions will result in lowering the liquid volume in the makeup tank and not result in an addition to the makeup tank. The probability of a moderator dilution accident does not increase due to the isolation of the makeup tank relief and vent pathway.

   Waste Gas Tank Rupture (14.2.2.7): Venting of the reactor coolant makeup tank is a contributor to the total activity contained in a WGDT (SAR 11.1.3.6.2). This activity will isolate the venting capability of the makeup tank to the WGDTs. The WGDT analysis is bounded by the maximum curie content of the WGDTs, which is analyzed and verified to not be exceeded throughout an operating cycle. Isolation of a source of radioactive gas will not result in an increase in the amount of waste gas collected by the WGDTs. Since the amount of gas collected by the WGDTs will not be increased, the probability of this accident is not increased.

   Therefore, the probability of an accident previously evaluated in the SAR will not be increased.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   The makeup tank vent and relief path isolation may impact the following SAR accidents:

   Waste gas tank rupture (14.2.2.7): Venting of the reactor coolant makeup tank is a contributor to the total activity contained in a WGDT (SAR 11.1.3.6.2).

   This activity will isolate the venting capability of the makeup tank to the WGDTs, therefore the consequences of this accident could be lessened

by the isolation of the makeup tank vent. The makeup tank vent is not credited with mitigating the consequences of a rupture of the waste gas tank. The consequences of an accident previously evaluated in the SAR will not, therefore be increased.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

The makeup tank is not safety related and is not required to be operable. The equipment important to safety which is associated with the makeup tank is the HPI system. Operability of the HPI system is ensured by maintaining the level and pressure in the makeup tank within procedural limits, or by isolating the makeup tank from the HPI suction header. The activity addressed by this procedure will make it impossible to vent the makeup tank in the event of an ECCS actuation, however, evaluations performed by ER 980331E101 and ER 980331E102 show that there is sufficient time to isolate the makeup tank from the HPI system to prevent the introduction of gas into the suction header. The control room operator will ensure that this action is taken in a timely manner.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

No equipment important to safety is directly affected by this activity except the HPI system as noted in the previous question. This activity will not have any effect on the operation of the HPI system as the makeup tank will be isolated from the HPI system in the event that the level in the makeup tank falls too low, and the control room operator will ensure that the tank is not overpressurized.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

The only effect that the isolation of the relief path could have on the makeup tank is to allow the pressure to increase beyond the relief valve setpoint or to cause the level to be lower than it otherwise would since the operator may divert letdown flow to reduce the pressure. If tank level decreases to 18", the tank will be isolated, preventing the introduction of air into the HPI suction header, and the HPI pumps will be aligned to the BWST. The overpressurization of the makeup tank is not possible with letdown diverted except by adding too much hydrogen. Since three operators are required to add hydrogen, and the fill rate is controlled by an operator in the control room who will be observing tank pressure, it is not credible that the tank could be pressurized enough to challenge the tank without operator intervention to prevent it. While the overpressurization of the makeup tank to the point at which it could rupture is not considered credible, the effects of this event have been investigated under EAR 92-003, which showed that the consequences of such an occurrence would have a less significant radiological release than the rupture of a waste gas decay tank. Furthermore, the original seismic category I design of the makeup tank has been maintained to ensure that boundary valve CV-1275 is protected during a DBE from seismic II over I hazards and possible hydrogen explosions caused by the rupture of a tank.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

As noted in the response to question 3, the only safety related equipment influenced by the makeup tank is the HPI system. The likelihood of a malfunction of the HPI system will not be increased by the isolation of the makeup tank vent path since the control room operator will isolate it from the HPI system if the level drops to 18". The HPI system has been evaluated in ER 980331E101 and ER 980331E102 to show that the system can be isolated in time to prevent the introduction of air into the suction header.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

No margin of safety has been identified in the basis for any technical specification which would be reduced by this activity. According to the bases of technical specification 3.1.4, one of the three ways that the activity resulting from a steam generator tube rupture could be brought back into specification is by venting the makeup tank gases. Only one of the three possible actions is required, however, and the other two actions, a gradual decrease in power or an increase in letdown rate, would still be available.

John Richardson — Certified Reviewer's Signature

John Richardson — Printed Name

2/12/2001 — Date

Reviewer's certification expiration date: 6/8/2002

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: [signature] Date: 2/22/01

46

PAGE 3 REV. 0 This Document contains 3 Pages.

Document No. ER 002475N101 Rev./Change No. 0

Title MSR DI Tie-Ins

Brief description of proposed change:

This change will install piping and valve tie-ins to allow installation of a future demineralization system for cleanup of a portion of the MSR belly drains from E12A and E12B to improve secondary chemistry control. Tie-ins are also provided to the condensate system for cycle heat recovery, to the demin. system for sluice water and compressed air, to ACW for cooling of MSR belly drains prior to demineralization, and to Condenser E11B to return the cleaned drain flow to the condensate system. All of these new connections will be provided with closed isolation valves and will serve as passive pressure boundary items until such time the MSR demineralization system is installed in a future NCP.

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report? Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7?

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions? (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. ER 002475N101 Rev./Change No. 0

Basis for Determination (Questions 1, 2, & 3):

Response to Question 1: Changes made here are below the level of detail and have no impact on these documents.

Response to Question 2: ANO-1 SAR Figures 9-9 and 10-2 are impacted by this design change as piping tie-ins are shown on these figures. An Evaluation is attached.

Response to Question 3: These changes involve no tests or experiments.

☐ Proposed change does not require 10CFR50.59 Evaluation per Attachment 1, Item #____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in questions 1, 2 and 3. If search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS: All	Index: 50.59 Unit 1 – moisture separator reheater, MSR, moisture separator, demineralizer*, condensate

MANUAL SECTIONS: 1SAR Chapter 10

FIGURES: 1SAR 9-9, 10-2

Certified Reviewer's Signature | David MacPhee Printed Name | 2/1/01 Date

Reviewer's certification expiration date: 9/16/01

Assistance provided by:

Printed Name	Scope of Assistance	Date

Search Scope Review Acceptability (NA, if performed by Technical Reviewer per 1000.006)

Certified Reviewer's Signature | EDWARD BLACKARD Printed Name | 2/1/01 Date

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

Document No. ER 002475N101 Rev./Change No. 0

Complete the following Determination. If the answer to any item below is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

FORM TITLE: 10CFR50.59 SAFETY EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 2 Pages.

Document No. ER 002475N101 Rev./Change No. 0 10CFR50.59 Eval. No. FRN#01-016 (Assigned by PSC)

Title MSR DI Tie-ins

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒
2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒
3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒
4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒
5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒
6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒
7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

Certified Reviewer's Signature	David MacPhee Printed Name	2/1/01 Date

Reviewer's certification expiration date: 9/16/01

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: _____ Date: 2/22/01

Document No. ER 002475N101 Rev./Change No. 0

10CFR50.59 Review Continuation Page

1. The probability of an accident previously evaluated in the SAR will not be increased.

The modifications made in this design change are incorporated in non-safety related system piping which do not act as accident initiators and whose failure will not initiate an accident. Thus the probability of analyzed accidents will not increase.

2. The consequences of an accident previously evaluated in the SAR will not be increased.

Systems modified by this change are not considered or credited in SAR accident analysis and thus the potential dose consequences of analyzed accidents are not affected.

3. The probability of a malfunction of equipment important to safety will not be increased.

The addition of passive valves and piping connection points to these existing non-safety related systems does not adversely affect the reliability of those systems nor introduce an additional mechanism for failure beyond that existing in the subject systems. The impact of failure of these modified systems upon safety related systems thus remains unchanged and the probability of malfunction in safety related systems is not changed.

4. The consequences of a malfunction of equipment important to safety will not be increased.

Changes made to these systems introduce no new failure modes which may affect safety related systems. Thus, the dose consequences of such non-safety related failure mode impacts on safety related systems wil not change.

5. The possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

These changes to existing systems do not introduce any new credible accident initiators nor introduce any new modes of failure not previously analyzed. Thus the possibility of an accident of a different type is not created.

6. The possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR will not be created.

This design change does not adversely impact the previously analyzed failure modes nor introduce any new failure modes for safety related equipment. Thus, the possibility of a malfunction of a different type is not created.

7. The margin of safety as defined in the basis for any technical specification will not be reduced.

Changes made here are below the level of detail and do not impact any margins of safety as defined in the Tech Spec bases. Thus the margin of safety as defined is not reduced.

47

FORM TITLE:	ARKANSAS NUCLEAR ONE	FORM NO.	REV.
	10CFR50.59 DETERMINATION	1000.131A	003-04-0

This Document contains 5 Pages.

Document No. 974078N101 Rev./Change No. 0

Title Repair of T-13 Retention Element

Brief description of proposed change: **This nuclear change will upgrade the spent resin tank internal components to prevent resin intrusion into the Liquid Radwaste and Gaseous Radwaste systems and to improve resin "fluffing" operation. The upgrades include the replacement of the sluice ring, the sparger ring, and the water outlet retention element. Additionally, piping modifications will be made which include new water sparger line with isolation valve CZ-105 and increasing the size of the spent resin tank drain line.**

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

ER974078N101
PAGE 4 REV 0

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

FORM TITLE:	ARKANSAS NUCLEAR ONE	FORM NO.	REV.
	10CFR50.59 DETERMINATION	1000.131A	003-04-0

Document No. **974078N101** Rev./Change No. **0**

Basis for Determination (Questions 1, 2 & 3):

Description

This modification will replace spent resin tank (T-13) internals including the sluice ring, air sparger ring, and the water outlet retention element. This change will prevent resin carryover which is a continuing problem throughout the liquid and gaseous radwaste systems and will also assist with resin "fluffing" operations. Additionally, piping modifications will be made which include a new water sparger line with isolation valve CZ-105 and increasing the size of the spent resin tank drain line and existing drain valve CZ-86.

Question 1

The spent resin tank internals and the configuration of the attached piping is below the level of detail contained in the Operating License documents.

Question 2

The spent resin tank is discussed in Section 11.1.3.3 of the Unit 1 SAR. However, the tank internals and attached piping is not addressed, so no SAR text is changed by this modification. Table 11-6 lists design information for the spent resin tank T-13, but this modification does not change any of these parameters. SAR Figure 11-1 Sh. 3 shows the tank and its attached piping. Piping internals are not shown on this drawing. This modification will relocate some attached piping to the tank which will require a change to the SAR Figure. Additionally, new valve CZ-105 is being added to isolate a new water sparger line. A 10CFR50.59 evaluation is required for this change.

Question 3

This modification does not address any test or experiment not discussed in the SAR.

☐ Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #_____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS:	
Unit 1 50.59	**("T-13", "spent w/3 resin", "resin w/3 tank", "sluice w/3 ring", "sparger" "LE-4622", "TE-4622")**
MANUAL SECTIONS:	
Unit 1 SAR	**11.1.3, Table 11-6**
FIGURES:	
Unit 1 SAR	**Figure 11-1 Sh. 3**

Stephen J. Lynn — Certified Reviewer's Signature

Stephen J. Lynn — Printed Name

2/20/01 — Date

Reviewer's certification expiration date: **5/26/01**

Assistance provided by:

Printed Name — Scope of Assistance — Date

ER974078N101
PAGE 6 REV 0

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

Certified Reviewer's Signature	Printed Name	Date
M. Keith Butler	M-Keith Butler	2-20-01

ENVIRONMENTAL IMPACT DETERMINATION
(UNIT 1 and UNIT 2)

Document No. **974078N101** Rev./Change No. **0**

Complete the following Determination. If the answer to any checklist item is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

FORM TITLE:	ARKANSAS NUCLEAR ONE	FORM NO.	REV.
	10CFR50.59 EVALUATION	1000.131B	003-04-0

This Document contains 2 Pages.

10CFR50.59 Eval. No. FRN#01-017
(Assigned by PSC)

Document No. **974078N101** Rev./Change No. **0**

Title **Repair of T-13 Retention Element**

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased?

Yes ☐ No ☒

This modification affects the Solid and Clean Liquid Radioactive Waste Systems. The replacement of the T-13 spent resin tank internals, the addition of a new water sparger line with isolation valve CZ-105, and increasing the size of the tank's drain line will not increase the probability of an analyzed accident since there are no accidents evaluated in the Unit 1 SAR related to these systems. These systems are not accident initiators and are not required for a safe shutdown of the plant. The purpose of this modification is to prevent resin carryover throughout various waste systems, to improve resin "fluffing" operations, and to decrease tank drainage time.

2. Will the consequences of an accident previously evaluated in the SAR be increased?

Yes ☐ No ☒

The new modifications to the spent resin tank internals and the addition of the new water sparger line will not increase the dose consequences of any analyzed accident. No radiological barriers are affected by this change and no new pathways for the release of radiation are created. Only existing penetrations on the spent resin tank will be used for these changes. The changes do not adversely affect any systems used to mitigate the consequences of an accident.

3. Will the probability of a malfunction of equipment important to safety be increased?

Yes ☐ No ☒

The installation of the new T-13 internals and the new water sparger line will not have any impact on any equipment important to safety. All components and materials associated with this change are non-safety related. The Solid and Clean Liquid Radwaste Systems serve no safety related function.

4. Will the consequences of a malfunction of equipment important to safety be increased?

ER974078N101
PAGE 9 REV 0

Yes ☐ No ☒

The installation of this modification can in no way affect offsite nor onsite dose consequences due to malfunctions of equipment important to safety. This modification only serves to prevent resin carryover from

the spent resin tank per the original design and provides an improved means of resin "fluffing" prior to transfer. The Solid or Clean Liquid Radwaste Systems are not used for any plant response to an analyzed accident. The dose for personnel responding to accidents can not be affected by the installation of this change and plant access is not affected.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created?

Yes ☐ No ☒

The only accident that could result from this modification is the leakage of liquid waste from the spent resin tank or attached piping and the carryover of spent resin. Leakage would be collected by area floor drains. Spent resin carryover throughout the waste systems is less likely due to the improved screen technology. There are no leaks postulated that could create doses in excess of 10CFR100 limits. All original piping codes and construction codes have been maintained, so no credible accident can be created by adding these modifications.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created?

Yes ☐ No ☒

These modifications to the spent resin tank internals and attached piping only affect the Solid and Clean Liquid Radwaste Systems and can in no way affect other equipment important to safety. There are no Seismic II/I concerns associated with this change. The new isolation valve CZ-105 will be normally closed and used for periodic resin "fluffing" using water from the T-12 clean waste receiver tanks.

7. Will the margin of safety as defined in the basis for any technical specification be reduced?

Yes ☐ No ☒

The Unit 1 Technical Specifications do not address any margins of safety for the Solid or the Clean Liquid Radwaste Systems.

Stephen J. Lynn — Certified Reviewer's Signature

Stephen J. Lynn — Printed Name

2/20/01 — Date

Reviewer's certification expiration date: 5/26/01

Assistance provided by:

Printed Name — Scope of Assistance — Date

ER974078N101
PAGE 10 REV 0

PSC review by: [signature] Date: 2/22/01

48

NCP 002612N101	ARKANSAS NUCLEAR ONE	
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. **ER 002612N101** Rev./Change No. **0**

Title **ANO-1 GL 96-06 Phase II Modifications**

Brief description of proposed change: Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," requested addressees to determine whether or not piping systems that penetrate containment are susceptible to thermal expansion of fluid so that over pressurization of piping could occur. Condition Report C-1996-0210 AI #13 reviewed ANO-1 for isolated and potentially isolated sections of piping inside the reactor building. The review identified seven (7) susceptible sections of piping that could potentially affect the integrity of reactor building penetrations. This modification will install three new thermal relief valves inside the ANO-1 reactor building. Three additional existing relief valves will be relocated from their current locations inside piping penetrations to sections of pipe further inside the reactor building. One existing redundant relief valve currently installed in a penetration will be removed from service. This modification will provide overpressure protection for six (6) of the potentially susceptible sections of piping identified in the CR. The seventh section of piping (between the letdown coolers and P14) is being separately evaluated for protection via administrative control.

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:
   QAPM? Yes☐ No☒
   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

Document No. **ER 002612N101** Rev./Change No. **0**

Basis for Determination (Questions 1, 2 & 3):

1. The ANO-1 Technical Specifications, Operating License, and Confirmatory Orders do not specifically address the issues associated with the potential for containment penetration overpressurization. There are no specific Technical Specifications or sections in the Operating License or Confirmatory Orders related to pressures in the piping through the subject containment penetrations. Adding the thermal overpressure protection does not affect the operation of the parent systems or containment isolation functions.

2. The change affects the ANO-1 SAR in that the SAR Figures, as noted, will require revision as a result of the proposed modification.

3. Thermal Relief valves are inspected and tested in accordance with Section XI of the ASME Boiler and Pressure Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a (g) (6) (i). Installing these relief valves does not constitute a test or experiment not described in the SAR.

☐ **Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #___. (If checked, note appropriate item #, send LDCR to Licensing).**

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS:	
LRS Search	(Penetration*, Overpressure W/10 Penetration, Containment, Relief, Thermal W/10 Relief, Relief W/10 Fire, Isolat*, Flange* W/20 Penetration, Leakage, Reactor Building, Containment Maintenance, Liner, Flange W/5 Leakage, GDC
MANUAL SECTIONS:	
1.2.4	Containment System
1.4.12	Criterion 16 - Containment Design
1.4.47	Criterion 54 - Piping Systems Penetrating Containment
1.4.49	Criterion 56 - Primary Containment Isolation
5.2.2	Design, Construction, and Testing of Penetrations
5.2.5	Isolation System
14	Safety Analysis
	FFN # 99-073 Safety Evaluation for DCP 97-4813D101
FIGURES:	
4-1	P&ID: Reactor Coolant System
7-20	P&ID: Reactor Coolant System
7-22	P&ID: Steam Generator Secondary System
9-7	P&ID: Intermediate Cooling System
9-12	P&ID: Decay Heat Removal
TABLES:	
5-1	Reactor Building Isolation Valves

William R Rowlett Jr — Certified Reviewer's Signature
William R. Rowlett, Jr. — Printed Name
12/11/00 — Date

Reviewer's certification expiration date: 05-25-01

Assistance provided by:

Printed Name	Scope of Assistance	Date
Kevin Broglie	Research	12/7/00

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

Alan Cox — Certified Reviewer's Signature
Alan Cox — Printed Name
12/11/00 — Date

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

Document No. ER 002612N101 Rev./Change No. 0

Complete the following Determination. If the answer to any checklist item is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower, which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge, which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area, which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

10CFR50.59 Eval. No. EE#01-018
(Assigned by PSC)

Document No. ER 002612N101 Rev./Change No. 0

Title ANO-1 GL 96-06 Phase II Modifications

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

Brief description of proposed change:

Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," requested addressees to determine whether or not piping systems that penetrate containment are susceptible to thermal expansion of fluid so that over pressurization of piping could occur. Condition Report C-1996-0210 AI #13 reviewed ANO-1 for isolated and potentially isolated sections of piping inside the reactor building. The review identified seven (7) susceptible sections of piping that could potentially affect the integrity of reactor building penetrations. This modification will install three new thermal relief valves inside the ANO-1 reactor building. Three additional existing relief valves will be relocated from their current locations inside piping penetrations to sections of pipe further inside the reactor building. One existing redundant relief valve currently installed in a penetration will be removed from service. This modification will provide overpressure protection for six (6) of the potentially susceptible sections of piping identified in the CR. The seventh section of piping (between the letdown coolers and P14) is being separately evaluated.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased?

Yes ☐ No ☒

The SAR Sections noted in the search scope of this evaluation were reviewed. The SAR does not discuss the overpressurization of isolated containment penetrations. The operability and design requirements of the containment isolation valves as addressed in the SAR are not affected by this change. The addition of the relief valves at the locations identified does not affect the operation of the parent systems or adjacent components. Consequently, the functionality of the affected systems is not changed.

The relief valves and their associated piping are designed in accordance with the same specified design requirements and design specifications as the existing equipment and components in the piping systems which will receive these relief valves.

Relief valve setpoints have been selected to prevent the inadvertent opening of these valves during normal system operational transients. There are no operator actions required to activate these valves. The thermal relief valves automatically open in response to an increase in the fluid pressure. Leakage of the relief valves will be identified via the containment sump leakage detection system. All leakage paths lead to the waste processing system for subsequent treatment.

Addition of the subject relief valves does not affect the performance of the parent fluid systems or the containment isolation system. The relief valves are added to protect the penetration and piping from thermal overpressure conditions postulated to occur during Infrequent Incidents (Emergency Conditions) or Limiting Faults (Faulted Conditions) when the associated penetrations are isolated and exposed to elevated ambient containment atmospheric temperatures as a result of a postulated accident.

Therefore, the addition of relief valves to the affected piping systems, to limit the over pressurization of the piping between isolated inboard/outboard containment isolation valves and isolated interior containment piping, will not increase the probability of any accident previously analyzed in the SAR.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

The addition of the thermal relief valves to the containment penetrations and isolated interior containment piping does not alter the functional or operational aspects of the piping systems. The thermal relief valves are located inside the containment building and discharge to the containment atmosphere. Over pressure will only occur in these penetrations/piping if both of the containment isolation valves are closed and trap fluid within the pipe, which is then heated up due to pipe exposure to elevated containment atmospheric temperatures. In this accident scenario, the relief valves provide no path for escape of radioactive fluids from the primary containment atmosphere. In the event of a relief valve failing open a design enhancement limits the flow through a nominal 1/16" orifice. A failure of a relief valve in the open position will not adversely impact boron concentration (sump dilution)

Consequently, the proposed modification does not affect the offsite dose to the public and thus, does not increase the consequences of any accident previously evaluated in the SAR.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

The relief valves and piping, added by this proposed modification, are specified to be in accordance with the system design parameters of the individual systems. Set pressures for the relief valves are chosen to protect the most limiting component within the pressure boundary of the parent containment penetration including piping, isolation valves, flued head assembly, and appurtenances.

The relief valves provide overpressurization protection to prevent the penetration lines from exceeding the ASME Code allowable stress limits. Their inclusion in the design will not adversely affect the operation and functionality of the containment isolation function for each process system included in this modification.

Addition of the relief valves does not affect the normal/accident function of the parent systems or negatively impact the containment isolation function. These valves do not have a negative impact on the previously installed equipment and do not increase the probability of any equipment or system malfunction. Valve testing will be performed in accordance with previously established methodologies for relief valves.

Based on this evaluation, the proposed modifications will not increase the probability of a malfunction of equipment important to safety.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

The operation (actuation) of the thermal relief valves only occur as a result of the penetration and/or piping being completely isolated (with virtually no valve leakage) by the containment isolation valves in response to an Infrequent Incident (Emergency Condition) or Limiting Fault (Faulted Condition). Effluent discharge from the relief valves will be collected by the plant drain systems and subsequently processed by the liquid waste processing system. The effluent volume is limited to a negligible fraction of the system volume between the containment isolation valves. As these valves are provided for thermal overpressure protection, their actuation will be intermittent and limited to the time required to relieve the excess pressure trapped between the closed containment isolation valves.

The Addition of relief valves does not affect the normal/accident function of the parent systems or negatively impact the containment isolation function. These valves do not increase the consequences of any equipment or system malfunction. These relief valves are designed to actuate in response to an Infrequent Incident (Emergency Condition) or Limiting Fault (Faulted Condition). The relief valves do not change the operational or performance characteristics of any equipment important to safety or preclude the necessary operation of any equipment important to safety.

Based on this evaluation, the proposed modifications will not increase the consequences of a malfunction of equipment important to safety.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

A review of the ANO-1 Accident Analysis, SAR Chapter 14, has been performed. The containment penetrations are isolated whenever ESAS actuation occurs. High Reactor Building pressure and in some cases low RCS pressure close the containment isolation valves. The addition of the thermal relief valves to the penetration piping does not change the operation or function of the isolation of the penetration piping or the Reactor Building. The actuation of the penetration thermal relief valves will occur as the result of an Infrequent Incident or Limiting Fault (small or large break LOCA, steam or feedwater line rupture, etc.). Uniform heating of the Reactor Building including the penetration piping is assumed to occur during this long-term event. Inboard and outboard containment isolation valves are provided to ensure that the Reactor Building may be isolated in the event that one of the isolation valves fails to close. Failure of a containment isolation valve to close or the failure of a penetration thermal relief valve to reseat does not breach the containment boundary.

The penetration and containment isolation valves function to mitigate an accident. The addition of the relief valves will not impact operation of the containment isolation valves. Based on this discussion, this change does not create the possibility of an accident of a different type than previously evaluated in the SAR.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

The addition of the thermal relief valves to the containment penetration piping system adds an additional potential leakage path from each of the process systems to which the valves are associated. The post accident failure mode for the thermal relief valves is the valve failure to reseat after actuation. Leakage from these valves would drain to the reactor building sump where the effluent is collected. Similarly, during normal operation, failure of these valves (either leaking or catastrophic failure) would be indicated by increased sump levels. During normal operation, operator action to identify the source of these leaks would be required. These failures are equivalent to failures of existing equipment important to safety.

Addition of the relief valves does not affect the normal function of the parent systems or negatively impact the containment isolation function. These valves do not have a negative impact on the previously installed equipment and do not increase the possibility of equipment or system malfunction. These relief valves are designed to actuate in response to Infrequent Incidents (Emergency Conditions) or Limiting Faults (Faulted Conditions). The relief valves do not change the operational or performance characteristics of equipment important to safety or preclude the necessary operation of equipment important to safety.

Installing the relief valves does not introduce any new piping or containment isolation failure mode beyond those previously evaluated in the SAR.

Consequently, the malfunction of equipment important to safety of a different type than any previously evaluated in the SAR has not been created by this modification.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

The Technical Specifications and their associated bases were reviewed for impact due to the changes made by this modification.

There are no Technical Specification Bases related to this change. The current Technical Specifications do not address the issues associated with the potential for containment overpressurization. There are no specific Technical Specifications related to pressures in the piping through the subject containment penetrations. Adding the thermal overpressure protection does not affect the operation of the parent systems or containment isolation functions. In addition, no margins of safety are considered by the Technical Specifications for this condition.

The offsite dose consequences will not be increased by the installation of the relief valves since the piping is isolated by ES actuation. There is no Technical Specification basis interpretation that applies to the configuration modification. Therefore, the margin of safety in TS bases is not reduced.

William R. Rowlett, Jr. — William R. Rowlett, Jr. — 12/11/00

Certified Reviewer's Signature | Printed Name | Date

Reviewer's certification expiration date: 05-25-01

Assistance provided by:

Printed Name	Scope of Assistance	Date
Kevin Broglie	Research	

PSC review by: R Tuttle Date: 3-1-01

49

FORM TITLE: ARKANSAS NUCLEAR ONE 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. **ER010182E101** This Document contains 4 Pages.
Rev./Change No. **0**

Title **Equivalency Evaluation for valve(s) DH-1016**

Brief description of proposed change: **Replace a safety-related, ASME, ¾ inch Globe Valve with an equivalent gate valve. Differences in the valve(s) have all been reconciled. Valve(s) conform to design bases.DH-1016 is an isolation valve for a system vent on the upstream side of the decay heat cooler E-35B.**

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. **ER010182E101** Rev./Change No. **0**

Basis for Determination (Questions 1, 2 & 3):
See continuation page for description of Bases.

☐ Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #, (If checked, note appropriate item #. send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS:	
50.59 Common	**DH-1016, GCB, E35B, vent w/10 decay heat**
U1 SER	**Low w/10 injection)**
U1 TRM	**Low w/10 injection)**
FHA	**Low w/10 injection, decay w/10 heat)**
E-plan	**Low w/10 injection, decay w/10 heat)**
U1 Confirm. Orders & LFO	**Low w/10 injection, decay w/10 heat)**
MANUAL SECTIONS:	
U1 & U2 Operating License	**All**
U1 SAR	**4.2.5.1,6.1.3.2, 9.5, A.7.6, Table 9-10, & 6-5.**
FIGURES:	
U1SAR	**Figure 9-12**

Murray C. Moser — Certified Reviewer's Signature
Murray C. Moser — Printed Name
03/01/01 — Date

Reviewer's certification expiration date: **8/04/01**

Assistance provided by:

Printed Name	Scope of Assistance	Date

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

William E. Rogers — Certified Reviewer's Signature
WILLIAM E. ROGERS — Printed Name
3/1/01 — Date

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

Document No. **ER010182E101** Rev./Change No. **0**

Complete the following Determination. If the answer to any checklist item is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

Document No. ER010182E101 Rev./Change No. 0

10CFR50.59 Review Continuation Page

BASES FOR RESPONSES TO DETERMINATION QUESTIONS 1:

A.) Will the proposed activity require a change to the Technical Specifications excluding the bases?

The plant modifications, which are the subject of this Determination, consist of the replacement of an existing valve(s) designated by component tag number DH-1016 by a proposed replacement valve.

The Technical Specifications describe safety limits, limiting conditions for operation, surveillance requirements, design features and administrative controls. With respect to valves, the related requirements of the Technical Specifications are the pressure relieving setpoints, surveillance and testing of valves and systems with valves, status of valves and their associated control circuits for certain activities or conditions, potential for valve leakage, and allowable isolation valve leakage rates. The replacement of an existing valve with an equivalent valve will not effect any of the requirements for valves contained in Technical Specifications. In addition, the specific valve(s) that is the subject of the Equivalency Evaluation is not mentioned in the Technical Specifications. The level of detail of the Technical Specification requirements allows the plant modifications, which are the subject of this determination to be implemented without requiring a change to the Technical Specifications.

B.) Will the proposed activity require a change to the operating license?

The operating license addresses the public health and safety, technical and financial qualifications, environmental, technical and other costs and benefits, maximum power level, physical protection, systems integrity, iodine monitoring, fire protection, and secondary water chemistry. With respect to valves the related requirements of the Operating License require a program to be implemented to reduce leaking from systems outside containment that would or could contain highly radioactive fluids during a transient or accident to as low as practical levels. The replacement of an existing valve with an equivalent valve will not alter or change the Operating License. The level of detail of the requirements of the operating license allow the plant modifications, which are the subject of this determination, to be implemented without requiring a change to the Operating License.

C.) Will the proposed activity require a change to the Confirmatory Orders?

Per review of the Confirmatory Orders issued to date, 1lfo0000.01 through 1lfo0000.14 and 2lfo0000.01 through 2lfo0000.08, there are no changes to the orders required due to the changes that are the subject of this determination.

BASES FOR RESPONSES TO DETERMINATION QUESTIONS 2:

The SAR documents were reviewed as indicated in the Search Scope Section of this Determination. Valve location, testing, closure time, environmental qualification, operation, status, position indication, seismic classification, failure to close, relief valve setpoints, conformance with GDC #55 and allowable leakage are discussed. The replacement of the existing valve with new valve that is equivalent with respect to the design bases requirements will not alter the description contained in the SAR documents. In addition the specific component tag number of the application considered in the evaluation is not mentioned in the text of the SAR documents. The SAR figure number 9-12 does show valve DH-1016. The existing valve is indicated to be a globe valve. The SAR figure will be revised to show a gate valve upon installation of the replacement gate valve that is the subject of the Equivalency Evaluation ER010182E101R0.

BASES FOR RESPONSES TO DETERMINATION QUESTIONS 3:

The proposed modifications do not involve a test or experiment.

FORM TITLE: ARKANSAS NUCLEAR ONE 10CFR50.59 SAFETY EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 1 Page.

Document No. ER010182E101 Rev./Change No. 0 10CFR50.59 Eval. No. FFN# 01-019 (Assigned by PSC)

Title Evaluate replacement for DH – 1016 Vavle.

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒
2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒
3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒
4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒
5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒
6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒
7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

Murray C. Moser — Certified Reviewer's Signature

Murray C. Moser — Printed Name

03/01/01 ~~02/20/01~~ 03/21/01 — Date

Reviewer's certification expiration date: 8/04/01

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: R.W. Fuller Date: 3-1-01

Document No. **ER010182E101R0** Rev./Change No. **0**

10CFR50.59 Review Continuation Page

Bases for responses to Safety Evaluation questions:

1.) Will the probability of an accident previously evaluated in the SAR be increased?

The replacement of a globe valve with a gate valve with all design bases characteristics of the replacement and existing valves being equivalent cannot increase the probability of any of the accidents evaluated in Chapter 14 of the U1 SAR. The valve is a normally closed valve and utilized to vent the system. The change in the valve disc style does not significantly affect any activity associated with this valve. The change in the valve's disc style from globe to gate is qualitatively assessed as not significantly changing the probability of an accident associated with any activity involving this valve.

2.) Will the consequences of an accident previously evaluated in the SAR be increased?

Radiation dose consequences are qualitatively assessed as not being increased by the change in the valve's disc style. The valve is located in the decay heat pump, P34B, room. The valve's leak rate and design bases pressure integrity are not significantly altered by the change in the valve disc style. The valve's size and operation remain the same. The LPI pumps are located in sealed rooms through which air does not circulate. Cooling is accomplished by a closed cycle ventilation system. Iodine leaking from this pump is not exhausted through the plant vent by the ventilation system. This valve replacement activity does not change, degrade or prevent actions that would be assumed or described in any accident scenario nor does it alter any assumptions that may have been made in evaluating the consequences of an accident. The valve replacement does not significantly affect any barriers that mitigate dose to the public or create a new pathway for release of radioactive material. The change in the valve disc style does not significantly effect onsite doses with respect to access to vital areas.

3.) Will the probability of a malfunction of equipment important to safety be increased?

The valve replacement activity does not degrade the performance of equipment important to safety below the design bases assumed by the ANO accident analysis for operation of the equipment. The change in the valve disc style does not significantly effect valve operation and all design bases requirements are satisfied by the replacement valve. The removal of decay heat and injection of borated water functions of the decay heat system will not incurr an increased probability of malfunction of equipment since all design bases for the valve are meet by the replacement valve.

4.) Will the consequences of a malfunction of equipment important to safety be increased?

Except for the valve disc style the existing and replacement valves are essentially like for like replacement with respect to the design bases and therefore would not increase the consequential effects of a malfunction of equipment. The normally open manually operated globe valve's failure position is assumed to be in the open position. If the failure position for the manually operated gate valve did change to closed position the activity would not result in an increase in onsite or offsite dose consequences of an accident.

5.) Will the possibility of an accident of a different type than any previously evaluated in the SAR be created?

The change in circumstances as a result of the replacement of the isolation valve which is currently a globe valve with a gate valve are not significant enough to alter any accident analysis or introduce any other type of accident. The replacement activity essentially involves a like for like replacement and therefore no additional unbounded types of accidents could be created by this activity.

ER010182E101R0 Document No. Rev./Change No. 0

10CFR50.59 Review Continuation Page

Bases for responses to Safety Evaluation questions:

6.) Will the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the SAR be created?

The change in circumstances as a result of the replacement of the isolation valve which is currently a globe valve with a gate valve are not significant enough to alter any accident analysis or introduce any other type of malfunction. The replacement activity essentially involves a like for like replacement and therefore no additional unbounded types of accidents or malfunctions are created. The replacement of a globe valve with a gate valve with all design bases characteristics of the replacement and existing valves being equivalent cannot introduce an initiator or failure not considered. The valve is a normally closed valve utilized to isolate a system vent. The change in the valve disc style does not significantly affect any activity associated with this valve. The change in the valve's disc style from globe to gate is qualitatively assessed as not significantly changing the possibility of a malfunction of equipment not previously evaluated.

7.) Will the margin of safety as defined in the basis for any technical specification be reduced?

There is no margin of safety involved in this activity. The replacement valve is an equivalent valve and does not create circumstances that could alter any margin of safety of the SSC.

50

FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 6 Pages.

Document No. CALC-00-R-1001-03 Rev./Change No. 0

Title CYCLE 17 RELOAD REPORT

Brief description of proposed change:

The Cycle 17 Reload Report provides the bases for the startup testing and operation of the Cycle 17 fuel cycle design. It is based on the results of safety analyses performed by Framatome Cogema Fuels (FCF). The design length of the fuel cycle is 526 EFPD, which includes an RCS T_{ave} reduction maneuver near the end of the cycle. The core design includes the insertion of fifty-six (56) FCF Mark B9ZL-NRLEF fresh fuel assemblies (Batch 19). Sixty (60) once-burned assemblies (Batch 18), and sixty (60) twice-burned assemblies (Batch 17) are shuffled to new core locations, with a center assembly from Batch 16A3. Additional details are discussed in the 50.59 Evaluation.

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report? Yes☒ No☐

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7?

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions? (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

Document No. CALC-00-R-1001-03 Rev./Change No. 0

Basis for Determination (Questions 1, 2, & 3):

See attached continuation page(s).

☐ Proposed change does not require 10CFR50.59 Evaluation per Attachment 1, Item # ____. (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in questions 1, 2 and 3. If search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document Section

LRS: See attached continuation page(s)

MANUAL SECTIONS: See attached continuation page(s)

FIGURES: See attached continuation page(s)

Darren M. Talley — Certified Reviewer's Signature

Darren G. Talley — Printed Name

5-19-01 — Date

Reviewer's certification expiration date: 12/7/02

Assistance provided by:

Printed Name	Scope of Assistance	Date
Larry Hu & David Smith	Core Design & Fuel Mechanical Design	1-29-01
Morris Byram	LOCA Analysis	1-25-01
Don Helm	Reactor Engineering / Startup Testing	1-25-01

Search Scope Review Acceptability (NA, if performed by Technical Reviewer per 1000.006)

Larry D. Young — Certified Reviewer's Signature

Larry D. Young — Printed Name

3/19/01 — Date

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

Document No. CALC-00-R-1001-03 Rev./Change No. 0

Complete the following Determination. If the answer to any item below is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

Document No. CALC-00-R-1001-03 Rev./Change No. 0

10CFR50.59 Review Continuation Page

Basis for Determination (Questions 1, 2, & 3)

Question 1: The Cycle 17 Reload Report describes and addresses the design, accident analyses, and limiting operating conditions for the ANO-1 Cycle 17 core. All cycle-specific technical specification limits and setpoints for operation of Cycle 17 are placed in the COLR as allowed by the NRC. TS 6.12.3.2 requires the use of the latest NRC approved Framatome Cogema Fuels Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses". All analysis methodologies used for Cycle 17 are approved by the NRC and referenced in BAW-10179P-A. Technical Specification safety limits, limiting safety settings, and limiting conditions of operation governing the operation of the unit are bounding for the Cycle 17 core. Therefore, no technical specification changes are required to support the startup testing and operation of Cycle 17 with regard to the Cycle 17 Reload Report.

The results of the reload analyses and the recommended operating limits and setpoints as stated in the Reload Report fall within the requirements for operating the ANO-1 core as referenced or described in the main body of the ANO-1 operating license. Therefore, no changes to the ANO-1 Operating License are required to support the operation of the Cycle 17 core.

In addition, the specific results of the analyses are beyond the scope of the Confirmatory Orders. Therefore, no changes to the ANO-1 Confirmatory Orders are required to support the operation of the Cycle 17 core.

Question 2: The Reload Report is intended to replace the contents of Chapter 3A of the SAR each cycle. Therefore, a SAR change is required for Chapter 3A and the Master Table of Contents. No other necessary changes to the remainder of the SAR have been identified. Likewise, the COLR must be updated to reflect the Reload Report limits and setpoints. The changes to the SAR and the COLR are described in the respective LDCRs being presented with this Reload Report 50.59 Review and will be addressed in the Evaluation. The change to the SAR Master Table of Contents is administrative in nature and does not change the scope of the SAR discussion. This particular change therefore meets exception F.2 of Attachment 1 of OP-1000.131 and will not be discussed in the Evaluation.

The specific results of the analyses and the recommended operating limits and setpoints as stated in the Reload Report and COLR are beyond the scope of the FHA. With regard to safe-shutdown capability, Cycle 17 calculations demonstrate that there will be sufficient RCS boration due to makeup for RCS shrinkage during cooldown. This fact is noted in the FHA with the Physics Manual listed as the reference.

The results of the reload analyses and the recommended operating limits as stated in the Reload Report and COLR fall within the requirements for operating the ANO-1 core as described in the bases to the technical specifications and do not result in invalidating any information presented in the ANO-1 Technical Specifications bases. Technical Specifications 3.1.4 and 3.10 bases describe dose calculations associated with the Steam Generator Tube Rupture, Main Steam Line Break, and Loss of Load events. These calculations were performed by the NRC to provide primary and secondary activity limits that result in exposures determined to be acceptable by the NRC. These calculations used assumptions that are different from those used in the Safety Analysis Report but the calculations were performed for reasons that are different, also. This reload report does not change any assumptions stated in the bases for LCOs associated with Technical Specifications 3.1.4 and 3.10.

The specific results of the analyses and the recommended operating limits and setpoints as stated in the Reload Report are beyond the level of detail present in the Technical Requirements Manual and do not result in invalidating any information presented in the Technical Requirements Manual.

The results of the reload analyses and the recommended operating limits and setpoints as stated in the Reload Report and COLR fall within the requirements for operating the ANO-1 core as described in the ANO-1 NRC Safety Evaluation Reports and do not result in invalidating any information presented in the ANO-1 NRC SERs.

Question 3: The startup tests and their acceptance criteria for Cycle 16 are described in the current SAR Chapter 3A. The Reload Report for Cycle 17 will replace SAR Chapter 3A, and the Reload Report describes the startup tests and their acceptance criteria for Cycle 17. The startup tests for Cycle 17 are the same as those of Cycle 16. Therefore, the proposed change does not involve any test or experiment which has not been previously described in the SAR.

Search Scope

	Document	Section
LRS	UNIT 1 50.59	ALL (reload*, core* design*, fuel* design*, operat* strategy*, burnup, imbalance*, cycle* w/5 16, mtc*, moder* coef*, moder* temp*, fuel w/3 *press*, temperat* w/10 reduc*, ejec* w/10 rod*, bypass* w/10 flow*, quad* powe* tilt*, qpt*, incor* detect*, short* w/5 emit*, radial* w/2 peak*, pin* w/2 peak*, peak* w/2 fact*, power peak, dropped* w/2 rod*, .65, 0.65, energy deposition, (stainles* steel*) w/2 rod*, clad* strain*, LHR*, linear* heat* rate*, enrichment*, shutdown* margin*, rem, rcs* w/5 flow*, 1.09, 109, tube* w/5 plug*, pluggin*, rod w/10 worth)
MANUAL SECTIONS	UNIT1 SAR	Sect. 9.6, all sections Ch. 14
	UNIT 1 COLR	ALL
	UNIT 1 TS and BASES	2.1, 2.2, 3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.1.7, 3.1.8, 3.2, 3.5.2, 3.8, 3.10, 4.9, 5.3, 5.4
FIGURES	UNIT1 SAR	Fig. 9-57, all Ch. 14 figures
	UNIT 1 COLR	ALL
	UNIT 1 TS and BASES	Figs. 3.1.2-1, 3.1.2-2, 3.1.2-3, 3.2-1, 3.8.1, 3.8.2, 5.4-1

Summary of LDCRs Required by this 50.59 Determination

Unit 1 SAR: The Reload Report is intended to replace the contents of Chapter 3A of the SAR each cycle. Therefore, a SAR change is required for Chapter 3A and the Master Table of Contents.

Unit 1 COLR: The COLR must be updated to reflect the Reload Report limits and setpoints.

There are no new Technical Specification changes or other Licensing Basis Document changes required for Cycle 17 startup testing and operation based on the Cycle 17 Reload Report.

FORM TITLE: 10CFR50.59 SAFETY EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 9 Pages.

Document No. CALC-00-R-1001-03 Rev./Change No. 0 10CFR50.59 Eval. No. FFN# 01-022 (Assigned by PSC)

Title Cycle 17 Reload Report

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒
2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒
3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒
4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒
5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒
6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒
7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

Darren G. Talley — Certified Reviewer's Signature

Darren G. Talley — Printed Name

3-19-01 — Date

Reviewer's certification expiration date: 12/07/02

Assistance provided by:

Printed Name	Scope of Assistance	Date
Larry Hu & David Smith	Core Design & Fuel Mechanical	1-29-01
Morris Byram	LOCA Analysis	1-25-01
Don Helm	Reactor Engineering / Startup Testing	1-25-01

PSC review by: [signature] Date: 3/22/01

10CFR50.59 Review Continuation Page

Basis for Answers to the Evaluation Questions

1. *Will the probability of an accident previously evaluated in the SAR be increased?*

For the accidents evaluated in the SAR, the only events which have an initiator which could be affected by the reload core design presented in the Reload Report for Cycle 17 are (a) the Stuck In/Stuck Out/Dropped Rod event, (b) Fuel Loading Errors, and (c) the Fuel Handling Accident.

The Mark-B9ZL-NRLEF fuel assembly design is utilized exclusively in the Cycle 17 core design loading. The fuel assembly design parameters for Cycle 17 are presented in Table 3A-1 of the Reload Report. BAW-10179P-A (Ref. 5 of the Reload Report) fuel assembly design criteria include the requirements (1) that a path for control rod insertion is ensured even for an assembly with the maximum credible damage, including a Safe Shutdown Earthquake; (2) that the holddown springs be capable of maintaining fuel assembly contact with the lower support plate during normal operation; and (3) that guide tube buckling not be allowed during normal operation or any transient condition where control rod insertion is required by the safety analysis. The dimensions and position of the Mark-B9ZL guide tubes are unchanged. Testing and in-reactor surveillance of rod drop times for Mark-B9ZL fuel assemblies with optimized guide tubes have demonstrated drop times comparable to drop times in fuel assemblies with standard guide tubes. Also, any dimensional changes due to irradiation, such as assembly bow, will not be altered since no change in the guide tubes material has occurred and the increased burnup is well within the industry experience base. Adequate control rod cooling will continue to be provided. Fuel rod bow to the point of contact with the guide tube where guide tube deformation could occur will continue to be precluded. The control rod assembly will not to be able to be disengaged from the fuel assembly guide tubes during operations. Therefore, there is no expectation that the probability of a Stuck In/Stuck Out/Dropped Rod event will be increased by the employment of the Cycle 17 core design presented in the Reload Report.

The fuel assembly identification will continue to be prominently displayed on the upper end fitting for core loading verification prior to startup, and operating procedures require verification of the final core loading. Therefore, the probability of gross fuel assembly misplacement in the core due to the Batch 19 Mark-B9ZL assemblies is not increased.

These assemblies have the same structural cage as that previously used at ANO-1 and will be capable of withstanding the expected handling loads. These assemblies are compatible with the fuel handling equipment. The manner of handling these assemblies will be unchanged. The envelope of the new fuel is no different than that of the past. The mass of these assemblies is approximately the same as the Batch 18 fuel. Hence, the probability of a fuel handling accident is not increased.

Therefore, the probability of an accident previously evaluated in the SAR is not increased.

2. *Will the consequences of an accident previously evaluated in the SAR be increased?*

The transient response of the plant to the abnormalities and accident scenarios analyzed in SAR chapter 14 will not be altered by the implementation of the Cycle 17 reload core design. As such, all associated accident initiators and any single-failure equipment malfunction postulations remain valid with respect to their impact upon the accident analyses.

The four-pump coastdown and the locked rotor event do not result in dose consequences since departure from nucleate boiling and/or cladding failure is precluded. The revised inputs (e.g., new RCS flow rate assumption) result in minimum DNB ratios well above the acceptance criterion to preclude DNB. Therefore, no dose consequences are produced from these events.

Table 3A-6 of the Cycle 17 Reload Report documents the results of the dose calculations based on the Cycle 17 core design and compares them to Cycle 16 results and the SAR. This table is reproduced in the supplemental information (see below) along with the values specified in the NRC SERs for ANO-1 as being acceptable. The slight changes in the dose consequences of the accident analyses are related to the use of

Cycle 17 specific radionuclide sources calculated from the actual Cycle 17 core design and irradiation history. These slight changes are not a result of changes in dose release scenario assumptions dictated by the accident scenario and the associated plant response. This table demonstrates that although some of the predicted doses for the Cycle 17 core design have increased a small amount relative to Cycle 16, all of the doses remain well below the acceptable SER doses. Therefore, the consequences of accidents previously evaluated in the SAR are not increased.

3. *Will the probability of a malfunction of equipment important to safety be increased?*

Equipment important to safety which could be impacted by the Cycle 17 reload core design includes: (1) control rods and drive mechanisms, (2) axial power shaping rods and drive mechanisms, and (3) RCS safety-related instrumentation (e.g., in-core detectors, pressure transducers, RTDs, level sensors, etc.).

As noted in the response to Question 1, fuel assembly design criteria assure that the reload core design will not impact the proper function of the control rods, axial power shaping rods, or their drive mechanisms. Cycle 17 operational characteristics will be very similar to Cycle 16. Thus operating pressures, temperatures, neutron fluxes, etc., will remain within the design parameters for RCS safety-related instrumentation as in Cycle 16. Likewise, the continued use of past operating characteristics and parameters which are bounded by current safety analyses (see response to Question #4) maintains the plant response to abnormalities or accident within the parameters used as design bases for engineered safety features. Also, there are no changes to plant equipment or plant operations required for the Cycle 17 reload core design. Therefore, the probability of a malfunction of equipment important to safety is not increased.

4. *Will the consequences of a malfunction of equipment important to safety be increased?*

The Reload Report concludes that by the examination of Cycle 17 core thermal, thermal-hydraulic, and kinetics properties, this core reload will not adversely affect the ability to operate the ANO-1 plant safely during Cycle 17. Considering the previously-accepted design basis used in the SAR and subsequent cycles, the transient evaluation of Cycle 17 is considered to be bounded by previously accepted analyses. The key safety analysis parameters for Cycle 17 are bounded by the assumptions in the SAR analyses and/or subsequent cycle analyses. The new analyses for the four-pump coastdown and locked rotor events remain bounded by the SAR analyses results.

The transient response of the plant to the abnormalities and accident scenarios analyzed in SAR chapter 14 will not be altered by the implementation of the Cycle 17 reload core design. As such, all associated accident initiators and any single-failure equipment malfunction postulations remain valid with respect to their impact upon the accident analyses. Slight changes in the dose consequences of the accident analyses are related to the use of Cycle 17 specific radionuclide sources calculated from the actual Cycle 17 core design and irradiation history (see Supplemental Information). These slight changes are not a result of changes in dose release scenario assumptions dictated by the accident scenario and the associated plant response. Also, the doses remain well within ANO-1 NRC SER allowable limits. Therefore, the consequences of a malfunction of equipment important to safety will not be increased.

5. *Will the possibility of an accident of a different type than any previously evaluated in the SAR be created?*

As noted in the response to Question #4, the Reload Report concludes that by the examination of Cycle 17 core thermal, thermal-hydraulic, and kinetics properties, this core reload will not adversely affect the ability to operate the ANO-1 plant safely during Cycle 17. Considering the previously-accepted design basis used in the SAR and subsequent cycles, the transient evaluation of Cycle 17 is bounded by previously accepted analyses. The key safety analysis parameters for Cycle 17 are bounded by the assumptions in the SAR analyses and/or subsequent cycle analyses.

The new analyses for the four-pump coastdown and the locked rotor event do not significantly alter the overall progression of these events as described in the SAR. The revised inputs (e.g., new RCS flow rate assumption) result in minimum DNB ratios which are well above the acceptance criterion to preclude DNB, and are bounded by the DNB ratios discussed in the SAR.

In addition, there are no changes to plant equipment or plant operations required for the Cycle 17 reload core design, nor is any new equipment required to be installed. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR is not created.

6. *Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created?*

   As noted in the response to Question #4, the Reload Report concludes that by the examination of Cycle 17 core thermal, thermal-hydraulic, and kinetics properties, this core reload will not adversely affect the ability to operate the ANO-1 plant safely during Cycle 17. Considering the previously-accepted design basis used in the SAR and subsequent cycles, the transient evaluation of Cycle 17 is considered to be bounded by previously accepted analyses. The key safety analysis parameters for Cycle 17 are bounded by the assumptions in the SAR analyses and/or subsequent cycle analyses. The new analyses for the four-pump coastdown and locked rotor events remain bounded by the SAR analyses results.

   In addition, there are no changes to plant equipment or plant operations required for the Cycle 17 reload core design, nor is any new equipment required to be installed. Therefore, the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR is not created.

7. *Will the margin of safety as defined in the bases for any technical specification be reduced?*

   The areas in which margin is defined in the bases of the TSs and that margins could be affected by the Cycle 17 reload core design are noted and discussed in the following:

   Instrument Error Adjustments in RPS setpoints – TS Bases 2.1 and 2.3: The TS Bases note that calibration and instrumentation errors are accounted for in the power/imbalance/flow, RCS pressure, and RCS outlet temperature RPS setpoints. The power/imbalance/flow setpoints, the Variable Low-Pressure Temperature (VLPT) setpoints, the VLPT protective limits, and the Pressure-Temperature Setpoints are developed and proposed in the Reload Report for the Cycle 17 COLR. The results presented in the Reload Report do account for calibration and instrumentation errors as required by NRC-approved methodology BAW-10179P-A. Therefore, this particular margin of safety is not reduced.

   Quadrant Power Tilt Limits – TS Bases 3.5.2: The TS Bases note that the QPT limits, in conjunction with the control rod position setpoints in the COLR, ensure that design peak heat rate criteria are not exceeded during normal operation including the effects of potential fuel densification. The Reload Report follows the methodology of BAW-10179P-A (Ref. 5 of the Reload Report). BAW-10179P-A addresses fuel densification in the determination of power distribution peaking margins. QPT is also considered in the determination of these margins. Rod position limits protect these peaking margins. Therefore, this particular margin of safety is not reduced.

   Stuck Rod Condition for Shutdown Margin – TS Bases 3.5.2 and 4.9: TS Bases note that shutdown margin is determined by assuming the highest worth control rod remains in the full out position. The Reload Report states that, "The adequacy of the shutdown margin with Cycle 17 stuck rod worths is demonstrated in Table 3A-4. Rod position setpoints that ensure the minimum shutdown margin is preserved during power operation, including during the EOC T_{avg} reduction maneuver, are specified in section 3A.8." (Reload Report, Sect. 3A.5.1). The shutdown margin calculations presented in Table 3A-4 do include allowance for maximum stuck rod worth. Therefore, this particular margin of safety is not reduced.

   ECCS Power Peaking, Shutdown Margin, and Potential Ejected Rod Worth as Ensured by Control Rod and APSR Position Limits – TS Bases 3.5.2: TS Bases note that the rod position limits are based on the most limiting of ECCS power peaking, shutdown margin, and potential ejected rod worth. The minimum available rod worth provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod remains in the full out position. The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65 %Δk/k at rated power or 1.0 %Δk/k at hot zero power. The Reload Report states that, "Calculated ejected rod worths and their adherence to criteria were considered at all times in life and at all power levels in the development of the rod position setpoints presented in section 3A.8. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with Cycle 17 stuck rod worths is demonstrated in Table 3A-4. Rod position setpoints that ensure the minimum shutdown margin is preserved during power operation, including during the EOC T_{avg} reduction maneuver, are specified in section 3A.8." (Reload Report, Sect. 3A.5.1). The Reload Report also states that, "Based on the analysis and the COLR revisions provided in this report, the Final Acceptance Criteria ECCS limits will not be exceeded, nor will the thermal design criteria be violated." (Reload Report, Sect. 3A.8). Therefore, this particular margin of safety is not reduced.

Conservatisms Applied to Power Distribution Factors – TS Bases 3.5.2: TS Bases note the application of the following specific uncertainty factors for conservatism in the power distribution factors for the operational reactor power-imbalance envelope defined in the COLR: (a) nuclear uncertainty factors, (b) thermal calibration, (c) fuel densification effects, (d) hot rod manufacturing tolerance factors, and (e) fuel rod bowing. The Reload Report follows the methodology of BAW-10179P-A (Ref. 5 of the Reload Report). BAW-10179P-A addresses these factors in the determination of power distribution peaking margins. Therefore, this particular margin of safety is not reduced.

Dose Limits Associated with the MSLB, SGTR, and Loss of Load Events – TS Bases 3.1.4 and 3.10: TS Bases mention three accident dose consequences, and these are tabulated below:

Accident	Dose in TS Bases
Steam Generator Tube Rupture	1.5 Rem to thyroid
	0.5 Rem to Whole Body
Main Steam Line Break	<28 Rem to thyroid
Loss of Load	1.5 Rem to thyroid

TSs 3.1.4 and 3.10 Bases describe dose calculations which were performed by the NRC to provide primary and secondary activity limits that result in exposures determined acceptable by the NRC. These NRC calculations are separate and distinct from the accident analysis dose calculations which form the ANO-1 licensing basis. The Reload Report describes results for Cycle 17 evaluations of the accident analysis dose consequences as discussed in the SAR, and does not address the analyses performed by the NRC as described in TS Bases 3.1.4 and 3.10. As such, any comparison of the dose results in these TS bases with the results in the Reload Report is invalid. This Reload Report does not change any assumptions stated in the Bases for LCOs associated with TSs 3.1.4 and 3.10. Therefore, this particular margin of safety is not reduced.

Therefore, no margin of safety as defined in the bases of the Technical Specifications is reduced.

Conclusion

Based upon the negative responses to the seven evaluation questions, the reload, startup testing and operation of the Cycle 17 fuel cycle design as described in the Cycle 17 Reload Report does not introduce an unreviewed safety question.

Supplemental Information

The information presented in the Reload Report for Cycle 17 startup testing and operation (CALC-00-R-1001-03, Rev. 0) forms the basis for most of the answers to the evaluation questions. The Reload Report was developed, reviewed, and approved by Framatome Cogema Fuels (FCF). The review process included submittal of a draft to EOI/ANO for review comments and the resolution of those comments. In addition to the Reload Report, FCF also issued a Reload Technical Document (CALC-00-R-1001-02, Rev. 0) which provides supporting information and detail with respect to the Reload Report. The review process for the Reload Technical Document also included submittal of a draft to EOI/ANO for review comments and the resolution of those comments.

In addition to normal reload evaluation issues, the Cycle 17 reload design process addressed the issue of steam generator tube plugging. The reload safety analyses evaluations addressed the effects of up to 20% SG tube plugging on accident evaluations which are evaluated on a reload basis. Core thermal-hydraulic analyses assumed a reduction in RCS flow from 109% of design flow to 105% of design flow to account for tube plugging effects. Fuel cladding corrosion analysis assumed an RCS flow of 108% of design flow vs. 109% of design flow to account for tube plugging effects. Details of these considerations are discussed in the information provided below, and in the Reload Technical Document for Cycle 17.

LOCA analyses, however, were not modified for tube plugging effects due to limitations in the CRAFT2 LOCA analysis code. Therefore, a LOCA-related limit on SG tube plugging of ~1200 tubes in any one SG remains applicable to Cycle 17. Contingency analyses and the necessary changes to the Reload Technical Document, the Reload Report, and the ANO-1 Licensing basis, in general are being addressed outside of this Reload Report 50.59. Provided that minimum RCS flow requirements specified in Section 3A.9 of the Cycle 17 Reload Report, and that no more than ~1200 tubes are plugged in any one SG, this 50.59 remains valid with respect to LOCA analysis.

The ANO-1 Licensing Basis Documents do not address assumptions regarding SG tube plugging. Therefore, the fact that assumptions and evaluations have been made in various reload analyses to provide margin for expected effects of increased tube plugging will not be addressed in this 50.59.

The supplemental information presented below is intended (1) to emphasize some of the issues within the Reload Report which have particular bearing upon the responses in this 50.59 Evaluation, and (2) to present the conclusions and supporting information developed from the reviews of certain areas/issues by EOI/ANO Reload Team personnel. Information which is a direct quotation from the Reload Report is noted by quotation marks and the associated Reload Report section(s).

Fuel System Design

"All fuel assemblies in Cycle 17 are the Mark-B9ZL NR-LEF (zone loaded, non-removable lower end fitting) design....Batch 17, 18, and 19 fuel utilize an improved spacer grid restraint system which increases the ability of the fuel assembly to maintain grid positions relative to the instrument tube....All fuel rods in Cycle 17 are the B9 design.... The batch 17, 18, and 19 fuel rods have incorporated additional refinements in the upper plenum spring design." (Reload Report, Sect. 3A.4.1).

"...[I]t can be concluded that all the Cycle 17 fuel rods are acceptable in terms of creep collapse....The stress parameters for the Cycle 17 fuel rods are enveloped by a conservative B9 fuel rod stress analysis....Results from the oxide analysis show that the Cycle 17 fuel meets the oxide criterion....All fuel assemblies in the Cycle 17 core are thermally similar. The design of the batch 19 Mark-B9 assemblies is such that the thermal performance of this fuel is equivalent to the fuel design used in the remainder of the core....The presence of four stainless steel rods in three batch 17 fuel assemblies was considered in the thermal evaluation...The compatibility of all possible fuel-cladding-coolant-assembly interactions for batch 19 fuel assemblies is identical to those of present fuel assembles because no new materials were introduced." (Reload Report, Sects. 3A.4.2, 3A.4.3, and 3A.4.4).

A reduced RCS flow rate assumption of 108% of design flow (vs. 109%) was used to evaluate oxide buildup due to cladding corrosion (see Table 4.2.10 of the Cycle 17 Reload Technical Document). This is a conservative measure to build in margin against potential flow reductions from OTSG tube plugging. The predicted corrosion buildup with this new assumption still meets the 100 μm acceptance criterion (Table 4.2.10 of the Reload

Technical Document). At the beginning of the cycle, RCS flow is verified to meet the minimum flow criterion stated in Section 9 of the Reload Technical Document.

In addition to the continued use of an improved spacer grid restraint system and the upper plenum spring design refinements, the Batch 19 assemblies also include the following minor design changes: (1) laser-etched bar coding on the lower end of the fuel cladding, and (2) a modified instrument tube design which replaces a two-piece tube assembly with a single-piece tube with a slight I. D. taper over the upper 7" of the tube. Laser coding of the cladding has been performed by at least one other fuel vendor (GE). Given the low temperature and stresses at the lower end of the cladding, no performance impacts will result. With regard to the modified instrument tube design, no impact to the strength, nuclear performance, or vibrational characteristics of the tube is expected (VDR, "Mark-B Fuel Design Changes -- Instrument Tubes," BPD-00-611, 8/30/2000, EDC File QR-026-27).

The Mark-B9ZL NR-LEF fuel assemblies have been used for the last three cycles, and were likewise reviewed under previous cycle 50.59s.

Nuclear Design

The Cycle 17 core design includes the insertion of fifty-six FCF Mark B9ZL-NRLEF fresh fuel assemblies (Batch 19). Sixty once-burned assemblies (Batch 18), and sixty twice-burned assemblies (Batch 17) are shuffled to new core locations. A Batch 16A3 assembly is used in the center position. The Batch 19 assembly design fuel enrichment is 4.03 w/o U-235 (192 rods at 4.05 w/o U-235 and 16 rods at 3.75 w/o U-235). The design length of the fuel cycle is 526 EFPD, which includes an RCS T_{ave} reduction maneuver near the end of the cycle.

"...The differences in feed batch size, BPRA loading, shuffle pattern, and cycle length caused the changes in the physics parameters between Cycles 16 and 17. Calculated ejected rod worths and their adherence to criteria were considered at all times in life and at all power levels in the development of the rod position setpoints presented in section 3A.8. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with Cycle 17 stuck rod worths is demonstrated in Table 3A-4. Rod position setpoints that ensure the minimum shutdown margin is preserved during power operation, including during the EOC T_{avg} reduction maneuver, are specified in section 3A.8...." (Reload Report, Sect. 3A.5.1)

"The design changes for Cycle 17 consist of an increase in feed batch size and a longer cycle length. There are also asymmetries in the full core loading by burnup which are fully outlined in Table 3A-1a. The asymmetries are not significant to the nuclear calculations because of the excellent balance in eigenvalues achieved in the design. These changes were incorporated in the physics model. The use of stainless steel replacement rods was also evaluated and determined not to significantly impact core reactivity, stuck rod worth, or ejected rod worth....The calculational methods used to obtain the important nuclear design parameters for this cycle were the same as those used for Cycle 16, which is the reference cycle. The core design change did not affect the methods for defining the transient neutronic parameters and thus, changes to these calculational methods were not required." (Reload Report, Sect. 3A.5.3)

The impact of Cycle 17 fuel on the fuel storage criticality calculations has been evaluated by Entergy. The acceptability of loading of Batch 17 and previous fuel batches in the region 1 racks was confirmed in a criticality analysis performed for ANO (CEO-98/00032, "ANO-1 SFP Region 1 Criticality Analysis Results," F.H. Smith to J.G. Head and N. Mosher, EDC File QR-104-36, January 28, 1998). In addition, comparative calculations have been performed which confirm that the Batch 19 fuel is bounded by the design basis fuel of 4.10 wt% U-235 for all storage areas (CEO-99/00176, "Criticality Confirmation for ANO-1 Cycle 16 Fresh Fuel," F.H. Smith to J.G. Head, EDC File QR-104-37, July 1999). These comparative results demonstrate that the Cycle 17 fuel does not violate the assumptions of the criticality analysis for the spent fuel storage racks, the fresh fuel storage rack, and the temporary containment rack. The Cycle 17 reload fuel can be safely stored without restriction in the fresh fuel storage rack, the temporary containment rack, and in region 1 of the spent fuel rack. The unirradiated Cycle 17 reload fuel can be stored in region 2 of the spent fuel storage rack in the restricted checkerboard configuration.

Thermal-Hydraulic Design

The Cycle 17 core thermal-hydraulic design is based on the use of Statistical Core Design (SCD) methodology as approved by the NRC (BAW-10187P-A, Ref. 13 of the Reload Report). SCD employs the comparison of DNB analyses results to a Thermal Design Limit (TDL) of 1.40 which reserves margin with respect to the allowable Statistical Design Limit (SDL) of 1.32. This design margin is reserved to accommodate minor design changes or minor deviations from the reference core DNB analysis.

The Cycle 17 core contains 177 Mark-B9 fuel assemblies using bypass flow optimized guide tubes, 60 control rods, 8 APSRs, and 40 BPRAs. The reference DNB analysis considers an all Mark-B9 core, 68 CR/APSR components, and 48 BPRAs. The Cycle 17 design results in a calculated bypass flow of 5.37 % of full flow vs. the 5.31 % calculated in the reference analysis. The Reload Report concludes that the difference in bypass flow is negligible.

A reduced RCS flow rate of 105% of design flow (vs. 109%) was used in the DNB analysis for Cycle 17 (see Section 6 of the Cycle 17 Reload Technical Document). This was a conservative measure to build in margin against the potential flow reduction from OTSG tube plugging. At the beginning of the cycle, RCS flow is verified to meet the minimum flow criterion stated in Section 9 of the Reload Report.

"A Cycle 16 evaluation of loose parts in the RCS that resulted in a 4 DNB point penalty remains applicable to Cycle 17. The effects of the higher Cycle 17 bypass flow on DNB, relative to the reference analysis is negligible, and the DNB penalty due to the evaluation of loose parts in the RCS is offset by retained DNB margin (difference between the Thermal Design Limit and the Statistical Design Limit). The impact of the new grid restraint system, described in Section 3A.4.1, has been incorporated into the hydraulic modeling of the core for Cycle 17....The effects of the four stainless steel rods in three batch 17 fuel assemblies were considered in all thermal hydraulic analyses."

Accident and Transient Analysis (LOCA and Non-LOCA)

"Each SAR accident analysis has been examined with respect to changes in the Cycle 17 parameters to verify that the SAR analyses are bounding for Cycle 17 operation, and to ensure that thermal performance during anticipated transients and accident events is not degraded." (Reload Report, Sect. 3A.7).

"The key cycle-specific parameters for each of the events in chapter 14 of the ANO-1 SAR were reviewed. It was concluded that the non-LOCA safety analyses remain bounding for Cycle 17 operation." (Reload Report, Sect. 3A.7.2)

"All batches of fuel in the Cycle 17 core were reviewed and shown to be bounded by a generic LOCA analysis." (Reload Report, Sect. 3A.7.3.2).

"It is concluded by the examination of Cycle 17 core thermal, thermal-hydraulic, and kinetics properties that this core reload will not adversely affect the ability to operate the ANO-1 plant safely during Cycle 17. Considering the previously-accepted design basis used in the SAR and subsequent cycles, the transient evaluation of Cycle 17 is considered to be bounded by previously accepted analyses. The key safety analysis parameters for Cycle 17 are bounded by the assumptions in the SAR analyses and/or subsequent cycle analyses." (Reload Report, Sect. 3A.7.4).

Because of the new assumption for RCS flow rate of 105% of design flow, the analysis for the Four-Pump Coastdown (4PCD) and the Locked Rotor (LR) event were reanalyzed with revised inputs consistent with the new flow rate assumption and new instrument uncertainty values (Section 7.2.6 of the Reload Technical Document). The new analyses for these events still result in DNB ratios which meet the analysis acceptance criterion. The 4PCD analysis results in a minimum DNB ratio of 1.98. The Locked Rotor analysis results in a minimum DNBR of 1.716. Each of these values is well above the 1.40 design limit established by Framatome to protect the 1.32 correlation limit as determined using Statistical Core Design methodology (see BAW-10187P-A). The results of the analyses as presented in the SAR (Chapter 14) remain conservatively bounding, specifically 4PCD minimum DNB ratio is ~1.5, and the LR event minimum DNB ratio actually reaches the correlation limit (thus, the cladding temperature response was analyzed). Although these analyses have been performed with updated inputs, since the SAR results remain bounding, the description in Section 14.2.6 of the SAR does not need to be updated. The updated inputs and results are presented in detail in the Reload Technical Document for Cycle 17, and summarized in the Reload Report for Cycle 17.

In addition to the 4PCD and LR events, the Reload Technical Document and Reload Report also describe the new analysis for the four-to-two pump coastdown (4-2PCD). The new 4-2PCD event analysis also includes revised inputs (e.g., RCS flow of 105% vs. 109%, and instrument error values). However, the results of this new analysis show that the minimum DNB ratio is 1.80, which is well above the Framatome design limit of 1.40 (Section 6.8.3 of the Reload Technical Document). Although it is evaluated each cycle, the 4-2PCD event is not addressed in SAR Chapter 14. Therefore, the SAR does not need to be updated for this event.

Dose Assessment

"All of the Cycle 17 accident doses are based on radionuclide sources calculated from the actual Cycle 17 core design and irradiation history. Table 3A-6 shows a comparison of the SAR, Cycle 16, and Cycle 17 doses for the chapter 14 accidents that result in significant offsite doses." (Reload Report, Sect. 3A.7.1).

The table below presents the information found in Table 3A-6, and the allowable doses based on ANO-1 NRC SERs are also presented below for comparison.

Accident	Dose (REM) SAR	Cycle 16	Cycle 17	NRC SER (allowable)
Fuel Handling (Outside RB)				
2 hr EAB Thyroid	9.537	10.4	10.4	75
2 hr EAB Whole-body	0.261	0.151	0.151	6
Fuel Handling (Inside RB)				
2 hr EAB Thyroid	63.599	69.1	69.2	75
2 hr EAB Whole-body	0.27	0.202	0.202	6
Steam Line Break				
2 hr EAB Thyroid	1.6	1.77	1.78	30
2 hr EAB Whole-body	0.0	0.009	0.009	2.5
Steam Generator Tube Failure				
2 hr EAB Thyroid	4.64	7.26	7.27	30
2 hr EAB Whole-body	0.125	0.3	0.3	2.5
Control Rod Ejection Accident				
2 hr EAB Thyroid	6.266	7.16	7.21	75
2 hr EAB Whole-body	0.012	0.007	0.007	6
30 day LPZ Thyroid	5.025	5.75	5.78	75
30 day LPZ Whole-body	0.009	0.005	0.005	6
Loss of Coolant Accident				
2 hr EAB Thyroid	7.01	3.78	3.81	300
2 hr EAB Whole-body	0.0165	0.03	0.03	25
30 day LPZ Thyroid	2.66	1.90	1.91	300
30 day LPZ Whole-body	0.0106	0.02	0.02	25
Maximum Hypothetical Accident				
2 hr EAB Thyroid	148.68	153.3	153.6	300
2 hr EAB Whole-body	4.66	5.29	5.30	25
30 day LPZ Thyroid	52.38	70.1	70.2	300
30 day LPZ Whole-body	1.54	1.89	1.89	25

Proposed Modifications to the Core Operating Limits Report

"Normal operating limits for Cycle 17 are defined by the error-adjusted alarm setpoints shown in Figures 3A-9 through 3A-17. APSR insertion limits and setpoints are specified in Table 3A-11. Quadrant power tilt limits and setpoints are listed in Table 3A-12." (Reload Report, Sect. 3A.8).

The decrease in the RCS flow rate assumption contributed to the "narrowing" of the RPS Power/Imbalance/Flow setpoints. Refinements in the instrument error used to adjust the raw setpoints (see Section 6.8 of the Cycle 17 Reload Technical Document), however helped to offset some of the "narrowing." The actual instrument error values used are below the level of detail of the ANO-1 Licensing Basis Documents.

"The linear heat rate limits specified in section 3A.7.3 were conservatively adjusted to build in additional LOCA PCT margin to accommodate modified plant conditions and are shown in Figure 3A-18." (Reload Report, Sect. 3A.8).

LOCA linear heat rate (LHR) limits for ANO-1 are currently based on a CRAFT2 LOCA analysis methodology. CRAFT2 cannot be readily used to justify OTSG tube plugging greater than approximately 1200 plugged tubes in any one OTSG. In order to build in margin to cover for up to 20% tube plugging (as the quote above from the Reload Report mentions), the CRAFT2 LHR limits were conservatively adjusted based on insight from RELAP5-based LOCA analyses. This adjustment ensures that the LOCA LHR limits, the Operational Imbalance setpoints, and the Rod Insertion Limits will be valid for Cycle 17 operation, even if OTSG tube plugging were to exceed the 1200 tube CRAFT2 limit. In the event that the CRAFT2 limit is exceeded, the ANO-1 SAR documents would need to be revised to adopt the RELAP5-based LOCA methodology, but the reload setpoints and LHR limits would still be valid. This conservative adjustment is also discussed in Section 8 of the Cycle 17 Reload Technical Document.

Startup Program – Physics Testing

The Startup Program – Physics Testing presented in the Reload Report for Cycle 17 is the same as that currently in the SAR for Cycle 17. Some minor changes to correct inadvertent typographical errors are to be implemented via the LDCR for SAR Chapter 3A. These changes have no impact which would result in the Cycle 17 startup program being substantially different from that of Cycle 16.

Reactor Vessel Fluence

The Cycle 17 core has been designed to obtain a low neutron leakage configuration similar to Cycle 16. The Cycle 16 fuel cycle design was determined not to result in a significant increase in the neutron flux, and therefore the fluence, to any of the welds or plates in the reactor vessel. The fuel cycle design of Cycle 17 is very similar to Cycle 16's; therefore there should not be a problem in the total fluence accumulation at EOL. The EOL fluence value was used as a basis for the current pressure/temperature and LTOP limits in the Technical Specifications.

51

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 3 Pages.

Document No. ER 010220 E101 Rev./Change No. 0

Title Continued Use of H-3

Brief description of proposed change: This ER justifies Use of H-3 in a degraded condition for 1R16

Will the proposed Activity:

1. Require a change to the Operating License including:

Technical Specifications (excluding the bases)?	Yes☐ No☒
Operating License?	Yes☐ No☒
Confirmatory Orders?	Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

SAR (multi-volume set for each unit)?	Yes☐ No☒
Core Operating Limits Report	Yes☐ No☒
Fire Hazards Analysis?	Yes☐ No☒
Bases of the Technical Specifications?	Yes☐ No☒
Technical Requirements Manual?	Yes☐ No☒
NRC Safety Evaluation Reports?	Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:

QAPM?	Yes☐ No☒
E-Plan?	Yes☐ No☒

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. ER 010220 E101 Rev./Change No. 0

Basis for Determination (Questions 1, 2 & 3):
Please see attached.

☐ Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #_____, (If checked, note appropriate Item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.

Document	Section
LRS:	
Unit 1 50.59	AR("H-3", fuel w/20 damage, fuel w/20 load, fuel w/20 handl", fuel w/20 hoist, hoist w/20 setpoint)
MANUAL SECTIONS:	
Unit 1 Technical Specifications	3.8, Table 4.1-2
Unit 1 Safety Analysis Report	3.1.2.4.2.1, 9.6, 14.2.2.3
FIGURES:	

Certified Reviewer's Signature | Jonathan M. Ralston (Printed Name) | 3-14-01 (Date)

Reviewer's certification expiration date: 3-16-02

Assistance provided by:

Printed Name	Scope of Assistance	Date
N/A		

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

Certified Reviewer's Signature | Printed Name | Date

FORM TITLE: ARKANSAS NUCLEAR ONE 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

ENVIRONMENTAL IMPACT DETERMINATION
(UNIT 1 and UNIT 2)

Document No. ER 910220 E101 Rev./Change No. 0

Complete the following Determination. If the answer to any checklist item is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 REVIEW CONTINUATION PAGE	FORM NO. 1000.131C	REV. 003-04-0

This Document contains 1 Page.

Document No. ER 010220 E101 Rev./Change No. 0

10CFR50.59 Review Continuation Page

The Spent Fuel Handling Bridge (H-3) is exceeding the hoist load setpoints by as much as 400 pounds. This ER evaluation justifies the use of the equipment in its current condition as a last resort should trouble shooting efforts not determine and correct the problem prior to using the equipment for 1R16.

There are three hoist setpoints on the H-3 bridge: overload, underload, and low load. The purpose of these setpoints is to preclude damaging fuel while handling. The overload/underload setpoint provides protection of the fuel assembly grid straps. The Low Load setpoint prevents excess weight from being placed on a seated fuel assembly by the fuel handling mast and grapple. As stated in the evaluation, the low load setpoint can be adjusted conservatively to meet the fuel vendor's requirement for maximum load on an assembly. Unfortunately, there may not enough margin for a setpoint change to preclude exceeding the vendor recommended overload/underload setpoints while still allowing operation of H-3 for movement of fuel.

Basis for Determination:

1. This ER Evaluation allows the use of the H-3 bridge for 1R16 in the condition described above. The Technical Specifications contain requirements for Fuel Loading and Refueling (Section 3.8), but setpoints for the fuel handling equipment is beyond the scope of this section and the remaining Operating License Documents. Table 4.1-2 Item 5 requires the refueling system interlocks be tested and functioning at the start of each refueling shutdown. This problem is effectively with the setpoint and all refueling interlocks are functioning. No detail or requirements are included for refueling machine setpoints. This ER evaluation will not require a change to these documents.

2. The Unit 1 SAR discusses the Fuel Handling System. The discussion of the Fuel Storage Handling Bridge is limited to its location and where it moves fuel to and from (Section 9.6.1.6). Section 3.3.3.3.2.1 discusses analyzed loads on fuel assemblies for LOCA and Seismic conditions. This analysis is limited to assemblies in the core and does not discuss handling loads. Section 14.2.2.3 presents the Fuel Handling Accident analysis. This analysis is initiated by a dropped fuel assembly causing damage to 82 fuel pins. Damage that could occur from exceeding the load setpoint by 400 pounds will be limited to the grid straps potentially rendering the assembly unusable in the core, but would not compromise cladding integrity. As discussed above, the condition evaluated by this ER evaluation is beyond the level of detail presented in the SAR documents. Therefore, this will not result in the SAR documents being no longer true or accurate, nor will it violate a requirement of the SAR documents.

3. This ER Evaluation allows the use of the H-3 bridge with hoist limits in a degraded condition. This does not constitute a test or experiment.

Although this resulted in a negative determination, the potential for damaging fuel warrants a discussion of the safety implications. Please see the attached 50.59 evaluation.

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 2 Pages.

10CFR50.59 Eval. No. FFN# 01-023
(Assigned by PSC)

Document No. ER 010220 E101 Rev./Change No. 0

Title Continued Use of H-3

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   Please See Attached

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   Please See Attached

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   Please See Attached

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   Please See Attached

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   Please See Attached

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   Please See Attached

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

Please See Attached

Certified Reviewer's Signature	Printed Name	Date
	Jonathan M. Ralston	3-14-01

Reviewer's certification expiration date: 3-16-02

Assistance provided by:

Printed Name	Scope of Assistance	Date
N/A		

PSC review by: _____ Date: 3/15/01

This Document contains 1 Page.

Document No. ER 010220 E181 Rev./Change No. 0

10CFR50.59 Review Continuation Page

As discussed in the ER Evaluation as well as the 50.59 determination above, these hoist setpoints are set to protect fuel assembly grid straps from damage. Damaged grid straps are not a safety question as long as a damaged assembly is evaluated prior to use in the core. Inspection of any assemblies experiencing an overload/underload in the spent fuel pool shall be performed prior to reloading the Cycle 17 core is a requirement stated in this ER Evaluation.

1. The fuel handling accident as discussed in section 14.2.2.3 of the SAR, is initiated by dropping a fuel assembly causing 82 pins of that assembly to be breached. These setpoints are for grid strap protection and not the protection of the fuel handling equipment which is designed to withstand loads far in excess of setpoint overrun that is occurring. The criticality analysis also examines a dropped fuel assembly with the most limiting case being an assembly placed between the racks and the pool walls. The analysis credits the Technical Specification required 1600 ppm and demonstrates that there are no consequences. Again, to initiate this accident, a fuel assembly must be dropped. Therefore continued use would not increase the probability of the occurrence of a fuel handling accident.

2. The consequences of a fuel handling accident do not change as a result of continuing the use of the Spent Fuel Bridge. This will have no affect on the spent fuel pool water level, fission product inventory, or any other assumption of the fuel handling accident analysis that could effect offsite dose consequences. The criticality analysis is not affected since this does not change the boron concentration of the pool or the geometry of the spent fuel racks.

3. Although the potential to damage the grid straps of assemblies will exist, the requirement of a visual inspection of any assemblies experiencing an overload/underload will preclude the use of an assembly having non acceptable damage in the core. There is no concern of damaging the spent fuel bridge as it is designed to handle loads far in excess of those observed as a result of this condition.

4. The consequences of a fuel handling accident are discussed in the SAR and are a function of the assumptions made in the analysis (water level, fission product inventory, number of failed pins, etc.). Continued use of the Spent Fuel Handling bridge does not impact any of these assumptions and will therefore not affect the consequences of a fuel handling accident. The criticality analysis is not affected as the boron concentration of the pool will not change, nor will the geometry of the spent fuel racks.

5. The only credible accident with respect to this condition is a fuel handling accident (dropped assembly). The criticality analysis also examines the dropped assembly and demonstrates there are no consequences. These are both analysed in the SAR and continued use of the equipment cannot lead to any other postulated accident.

6. The only equipment involved in this evaluation is the spent fuel handling bridge, fuel, and the spent fuel racks. A fuel handling accident (a fuel assembly dropped from the fuel handling equipment) is analyzed in the SAR along with the fuel storage rack accident conditions with respect to the criticality analysis (dropped fuel assembly). No other equipment is involved.

7. No Technical Specification Bases exist with respect to fuel handling system setpoints.

E-DOC TITLE:	E-DOC NO.	CHANGE NO.
50.59 REVIEW FORM	LI-101 Att 9.1	1

Facility: ANO - Unit 1 **This Document Contains** 4 **Pages**

Document Reviewed: ER010220E101

System Designator(s): EH

Check the applicable review(s):

☒	**SCREENING**	**Sections I, II, and III required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, III, and IV required**
☐	**50.59 EVALUATION** **Evaluation #:**	**Sections I, II, III, and V required**

NOTE: Only the sections required as indicated above must be included in the Review.

I. SIGNATURES / OVERVIEW

Preparer: [signature] / Jonathan M. Ralston / EOI / Nuc. Eng / 8-27-01
Signature / Name (print) / Company / Department / Date

Reviewer: [signature] / John Ellis / EOI / SYE-1 / 8-28-01
Signature / Name (print) / Company / Department / Date

(PSRC): N/A
Chairman's Signature / Date (N/A for Screenings and 50.59 Evaluation Exemptions)

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
Jay Wellwood	72.48 Review

Description of Proposed Change

Extends the revision 0 justification to including handling fuel for dry cask activities prior to 1R17.

II. SCREENING

A. Licensing Basis Document Review

Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?

Operating License	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒		
TS	☐	☒		
NRC Orders	☐	☒	☐	
If "YES", obtain NRC approval prior to implementing the change. (See Section 5.1.13 for exceptions.)				

LBDs controlled under 50.59	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
UFSAR	☐	☒		
TS Bases	☐	☒		
Technical Requirements Manual	☐	☒		
Core Operating Limits Report	☐	☒		
Fire Hazard Analysis	☐	☒	☐	
Fire Protection Program	☐	☐	☒	
Offsite Dose Calculations Manual	☐	☒	☐	
Process Control Program	☐	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒		
If "YES", perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.				

LBDs controlled under 72.48	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Cask UFSAR	☐	☒	☐	
Certificate of Compliance	☐	☒	☐	
If "YES", evaluate/process any changes in accordance with 72.48				

LBDs controlled under other regulations	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☒		
Emergency Plan[2]	☐	☒		
Security Plan[2, 3]	☐	☒		
Inservice Inspection Program[4]	☐	☐	☒	
Inservice Testing Program[4]	☐	☐	☒	
If "YES", evaluate/process any changes in accordance with the appropriate regulation.				

[1] If "YES," see Section 5.1.5.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed.
[3] The Security Plan is classified as safeguards and can only be reviewed by personnel with the appropriate security clearance. The Preparer should notify the security department of potential changes to the Security Plan.
[4] If "YES", process the change in accordance with the 10CFR50.55a control program.

B. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No — If "yes," perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.

C. **Basis**

(Provide a basis for the "no" items checked in Sections II.A and II.B, above. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. If a 50.59 Evaluation is required, this section may be N/A'd.)

This screening documents the review of FFN#01-023 performed for revision 0 of this Evaluation on March 14, 2001. The original 50.59 was reviewed per section 5.4.5 of LI-101 and continues to cover the proposed activities (fuel handling). A document search was performed (see section E below) and found that no subsequent changes have beem made to the licensing basis documents impacting the previous 50.59 review and evaluation. The original 50.59 is attached. In addition, because the continued fuel handling activities will include dry fuel, a 72.48 review is required and is attached.

D. **Is the validity of this Review dependent on any other change?** (See Section 5.2.2.4 of the EOI 10CFR50.59 Program Review Guidelines) ☐ Yes ☒ No

If "Yes," list the required changes.

E. **References**

[Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.2.2.4 of LI-101.]

Documents:	Keywords:
Unit 1 50.59	"H-3", fuel w/20 damage, fuel w/20 load, fuel w/20 handl*, fuel w/20 hoist, hoist w/20 setpoint
FSAR Sections Reviewed:	FSAR Figures Reviewed:
Unit 1 TS 3.8, Table 4.1-2	None
Unit 1 SAR 3.1.2.4.2.1, 9.6, 14.2.2.3	

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations."

Will the proposed Change being evaluated:

Yes	No	
☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
☐	☒	Involve dredging activities in a lake, river, pond, or stream?
☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
☐	☒	Discharge any chemicals new or different from that previously discharged?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?
☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?
☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
☐	☒	Involve the installation or modification of a stationary or mobile tank?
☐	☒	Involve the use or storage of oils or chemicals?
☐	☒	Involve burial or placement of any solid wastes in the site area that may effect runoff, surface water, or groundwater?

52

ATT. 3

FORM TITLE: ARKANSAS NUCLEAR ONE 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 4 Pages.

Document No. ER010118E101 Rev./Change No. 0

Title Provide Temporary Power to D-11 & D-21 in support of maintenance on 43-D01 & 43-D02 Transfer Switches

Brief description of proposed change: Provide Temporary Power to D-11 & D-21 in support of maintenance on 43-D01 & 43-D02

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☐ No☒

   Core Operating Limits Report Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ATT. 3

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. ER010118E101 Rev./Change No. 0

Basis for Determination (Questions 1, 2 & 3):

1. This change will not affect existing Tech Specs, Operating License or Confirmatory Orders. This change will only be allowed during Modes 5 (cold shutdown) or 6 (refueling). The change will continue to prevent both 125V DC buses from being tied together, similar to the "break before make" function of the transfer switch.
2. FSAR figure 8-1, identifies the transfer switches in the form of a parallel set of contacts that allow D11 to be powered from D01 or D02 and D21 to be powered from either bus D01 or D02. This ER will effectively jumper out this transfer switch to allow maintenance on the switch while maintaining D11 (D21) energized from its designed "emergency supply". No changes to existing design features that provide protection and isolation between the two redundant 125V DC buses will be altered by this change (emergency supply feeder cables, fuses, etc.). The design function of the transfer switch is to support the ability of maintaining power to D11 or D21 during their associated Division's Bus outage. Existing Operating procedures allows D11 or D21 from being powered from either Bus in modes 5 & 6, which are the only modes that installation of this ER will be allowed. SAR sections 8.3.2.1.3 & 8.3.2.1.4 briefly discuss these transfer switches and only identify that they are designed to prevent tying both Buses together simultaneously. The change authorized by ER010118E101, will continue to maintain isolation between the two 125V DC buses, thus not invalidating these SAR sections.
   Per attachment 1 to OP-1000.131, section B.6, a 10CFR50.59 evaluation is not required if the affected figure is associated with a temporary valve lineup. Since this change is temporary in nature, supplies the D11 and or D21 panels from their designed "emergency supply", similar to the transfer switch, then this change and its impact on SAR Figure 8-1 will not be evaluated via a 10CFR50.59 evaluation.
   No credit is taken for function of these switches during a postulated "Fuel Handling Accident".
3. This change will not involve a test or experiment.

☒ Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #_____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS:	
50.59 – Unit 1	**"Transfer Switch", "43-D01", "43-D02", "D11", "D21"**
MANUAL SECTIONS:	
8.3.2 – DC Power System;	
14.2.2.3 – Fuel Handling Accident	
FIGURES:	
8-1	

Brad Risner — Brad Risner — 8/16/01

Certified Reviewer's Signature — Printed Name — Date

Reviewer's certification expiration date: 5/2/02

ATT. 3

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Assistance provided by:

Printed Name	Scope of Assistance	Date
Adrian Meyer	Initial Draft and LRS Search	2/15/01

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

Certified Reviewer's Signature	Printed Name	Date
	John Ekin	2/16/01

ATT. 3

FORM TITLE: ARKANSAS NUCLEAR ONE 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

ENVIRONMENTAL IMPACT DETERMINATION
(UNIT 1 and UNIT 2)

Document No. ER010118E101 Rev./Change No. 0

Complete the following Determination. If the answer to any checklist item is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

ATT. 3

ARKANSAS NUCLEAR ONE			Page 1
FORM TITLE: 3/24/01 10CFR50.59 SAFETY EVALUATION		FORM NO. 1000.131B	REV. 003-04-0

ALM 1412.116-003-03-0

Document No. ER010118E101 Rev./Change No. 0 This Document contains 4 Pages.
10CFR50.59 Eval. No. FFN #01-024 (Assigned by PSC)

Title Provide Emergency Power to Distribution Panels (D11 & D21) in support of Preventative Maintenance on transfer switches 43-D01 & 43-D02.

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

Description of Change:

The subject ER provides an evaluation for installing jumpers from the designed "Emergency Supply" to the associated distribution panels' feeder cables (D11 or D21). This change will support planed preventative maintenance activities on the D11/D21 distribution panel transfer switches (43-D01 & 43-D02). The transfer switches (43-D01 & 43-D02) provide a means for Operations to supply D11 or D21 from their redundant 125V DC buses. Procedure controls (OP-1107.004, Section 8.0) are in place that limit the time and required plant conditions when a redundant bus (D01/D02) is allowed to supply the redundant distribution panel (D11/D21). The procedurally controlled T-Alt's "Plant Impact Statement" refers to OP-1107.004, section 8.0 under "Plant Mode" for guidance on when the T-Alt may be implemented. The subject ER is in support of a procedurally controlled T-alt that will only allow installation of the T-alt when Unit 1 is in Mode 5 (Cold Shutdown) or Mode 6 (Refueling) to support required transfer switch preventative maintenance.

This evaluation is being performed due to SAR figure 8-1, showing a 1-line representation of the 43-D01 & 43-D02 transfer switches. Implementation of this T-alt will effectively jumper the associated transfer switches out of the circuit. However the load (D11 & D21) will continue to receive power from their designed "Emergency Supply".

The following Emergency Operating Procedures and Abnormal Operating Procedures were reviewed to determine if the subject transfer switches were relied upon by Operations to mitigate these events. This review determined that the subject transfer switches are not relied upon to mitigate these events. A word search utilizing "43" and "transfer switch" was used to support this search.

OP-1202.007: "Degraded Power"

OP-1202.008: "Blackout"

OP-1203.002: "Alternate Shutdown"

OP-1203.020: "Load Reject"

OP-1203.029: "Remote Shutdown"

OP-1203.037: "Abnormal ES Bus Voltage"

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

A review of "Identification of Cause" sections within sections 14.1 and 14.2 of the Unit 1 SAR, associated with each of the "Abnormalities and Accidents" evaluated was reviewed. This review did not identify the subject transfer switches (43-D01, 43-D02) distribution panels (D11, D21) or associated power sources (D01, D02) as an accident initiator. Since the equipment altered by this change is not an initiator to any of the evaluated "Abnormalities or Accidents" evaluated in Chapter 14 of the SAR, the proposed change will not increase the probability of an accident previously evaluated in the SAR.

ATT. 3

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

The proposed change will not adversely affect the performance of any barriers (Fuel Cladding, RCS or Containment) that are utilized to limit the consequences (offsite dose) of an accident. This change will not adversely affect the performance of any ECCS system in mitigating the consequences on an accident. Therefore, this change will not increase the consequences of any accident previously evaluated in the SAR.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

The proposed change will involve de-terminating jumpers and cables local to the associated transfer switch and installation of jumpers local to the transfer switch. The jumpers added by the proposed change shall be Safety Related – quality level L1. Adequate PMT steps are provided after both installation of the proposed T-alt and removal of this T-alt that performed the following:

a. Check the affected circuits to ensure that grounds are not introduced on the associated DC bus

b. Check the affected circuits to ensure that degradation to the cable/jumper insulation has not occurred and

c. Check that the cables are terminated properly for maintaining the proper polarity of the DC supply to the associated DC distribution panels.

The "Emergency Supply" feeder cables are fused to protect the "Emergency Supply Bus" (D01 & D02) in the event of a postulated fault on the "Emergency Supply" feeder cable. Likewise, all feeder circuits in both D11 and D21 distribution panels are protected with breakers for coordination between the associated loads and their supply bus.

This ER limits implementation of this T-Alt to modes 5 & 6, consistent with procedure OP-1104.007, (Battery & 125V DC Distribution). Procedure OP-1104.007, identifies the potential of a single active failure of causing a loss of both EDGs with the transfer switch aligned to its Emergency Supply. With this concern the procedure (OP-1107.004) limits this configuration (Emergency Supply to Distribution Panels) to "Operation in this mode is not allowed unless plant conditions warrant." The transfer switches maintenance procedure's (1409.116) Plant Impact Statement identifies OP-1107.004, section 8.0 as allowed "Plant Modes" for installation of this T-Alt.

Adequate material, PMT steps and circuit protect exist to ensure the proposed changes doe not increase the probability of a malfunction of equipment important to safety.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

The proposed change will install jumpers to provide power to distribution panels D11 (Red Train) or D21 (Green Train) from their redundant DC bus (D02 or D01 respectively). This change is similar to the associated transfer switch being aligned to its "Emergency Supply" with the exception that returning the load to its "Normal Supply" via the transfer switch will be disabled. These distribution panels provide the DC control power for starting and running their associated EDGs. A postulated loss of either DC bus, while supplying its redundant DC distribution panel (D11 or D21) would result in a shutdown of both EDGs and prevent a normal start of both EDGs. The EDGs would have to be started using guidance contained in OP-1104.036 "Starting the EDG without DC control power". The basis for this procedurally controlled T-Alt is to support maintaining the D11 or D21 distribution panels energized during their associated Train's maintenance window while preventative maintenance is performed on the associated Train's transfer switch.

ATT. 3

The Plant is not designed to be "Singe Failure Proof" during refueling outage safety train maintenance windows.

This ER limits implementation of this T-Alt to modes 5 & 6, consistent with procedure OP-1104.007, (Battery & 125V DC Distribution). Procedure OP-1104.007, identifies the potential of a single active failure of causing a loss of both EDGs with the transfer switch aligned to its Emergency Supply. With this concern the procedure (OP-1107.004) limits this configuration (Emergency Supply to Distribution Panels) to "Operation in this mode is not allowed unless plant conditions warrant." The transfer switches maintenance procedure's (1409.116) Plant Impact Statement identifies OP-1107.004, section 8.0 as allowed "Plant Modes" for installation of this T-Alt.

The "Abnormalities & Accidents" evaluated in SAR Chapter 14 are evaluated with a core thermal power of 2568MWt. However, the Moderation Dilution Accident, Fuel Handling Accident and Rod Ejection Accident discuss the affects/consequences of these accidents during shutdown conditions. The consequences of these accidents at full power operation bounds the consequences of these accidents during shutdown conditions. Existing Plant procedures (OP_1107.004) limit the times which the D11 or D21 distribution panels may be supplied from their "Emergency Supply". Implementation of this T-Alt will continue to be limited by OP-1107.004. Therefore this change does not increase the consequences of a malfunction of equipment important to safety.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

The proposed change is effectively aligning the designed "Emergency Supply" source from the vital DC buses (D01 & D02) to their respective redundant distribution panels (D21 & D11). This change is only removing the transfer switch that supports alignment of the "Emergency Supply" to the associated DC distribution panel for maintenance activities on the switch. Therefore, since the proposed change will not put the vital DC buses or their associated DC distribution panels outside of their existing "design basis" and the T-alt will only be allowed during Plant conditions that the condition is presently allowed, this change will not create the possibility of an accident of a different type than any previously evaluated in the SAR.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

The proposed change is effectively aligning the designed "Emergency Supply" source from the vital DC buses (D01 & D02) to their respective redundant distribution panels (D21 & D11). This change is only removing the transfer switch that supports alignment of the "Emergency Supply" to the associated DC distribution panel for maintenance activities on the switch. Therefore, since the proposed change will not put the vital DC buses or their associated DC distribution panels outside of their existing "design basis" and the T-alt will only be allowed during Plant conditions that the condition is presently allowed, this change will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

ATT. 3

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

The proposed change is effectively aligning the designed "Emergency Supply" source from the vital DC buses (D01 & D02) to their respective redundant distribution panels (D21 & D11). This change is only removing the transfer switch that supports alignment of the "Emergency Supply" to the associated DC distribution panel for maintenance activities on the switch. Therefore, since the proposed change will not put the vital DC buses or their associated DC distribution panels outside of their existing "design basis" and the T-alt will only be allowed during Plant conditions that the condition is presently allowed, this change will not reduce the margin of safety as defined in the basis for any technical specification.

Certified Reviewer's Signature | ADRIAN MEYER Printed Name | 3/20/01 Date

Reviewer's certification expiration date: 3/14/03

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: _____ Date: 3/24/01

53

FFN # 01-025

ARKANSAS NUCLEAR ONE		Page 1
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 3 Pages.

DM 3/27/01

TAP 01-1-002

Document No. ~~ER010269E101~~ Rev./Change No. 0

Title Evaluate Operation with Valve MU-45A Internals Removed

Brief description of proposed change:

Valve MU-45A is the A HPI nozzle isolation valve between the RCS and HPI system. It is used to isolate these two systems only while the unit is shut down. During maintenance the valve was found with damage to its disc and in body seats. The in-body seats and the discs (double disc) were damaged beyond repair. Because of the size and location of the valve, it must be cut out of the line in order to make repairs of this magnitude. Since the valve has no active function during normal power operations or during an accident, it was decided to remove the valve disc and leave the valve in place without the disc. The RCS to HPI isolation boundary will be established at the HPI injection valves (which are normally closed) when needed.

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report? Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7?

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions? (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

TAP 01-1-002

Document No. ~~ER010269E101~~ Rev./Change No. 0

10CFR50.59 Review Continuation Page

MU-45A need only to maintain the RCS pressure boundary during normal or EOP operation. Removing the valve disc will not change its function and therefore, this change will have no adverse effect on the RCS or HPI systems during normal or EOP conditions. P&ID M-230 sheet 1 shows the valve in the line in a normally open position which is correct for normal power operations and EOP conditions. Since the valve disc will be removed, the valve essentially becomes a section of pipe with no capability of isolation. The P&ID will be changed to note that this valve has no disc installed and cannot be used for isolation. This is a SAR figure and therefore an evaluation has been completed.

The valve is normally open during power operations and EOP conditions. Removing the disc does not change the normal operating position or function of the valve at power operations or EOP conditions since the requirement is for this valve open. In other words, the valve is open and will continue to be open without the disc, therefore, this does not involve a test or experiment not described in the SAR.

FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

TAP 01-1-002

DM 3/27/01

Document No. ~~ER-010269E101~~ Rev./Change No. 0

Basis for Determination (Questions 1, 2, & 3):

Valve MU-45A is the A HPI nozzle isolation valve between the RCS and HPI system. It is used to isolate these two systems only while the unit is shut down. During maintenance the valve was found with damage to its internals. The in-body seats and the discs (double disc valve) were damaged beyond repair. Because of the size and location of the valve, it must be cut out of the line in order to make repairs of this magnitude. Since the valve has no active function during normal power operations or during an accident, it was decided to remove the valve internals and reestablish the RCS to HPI isolation boundary at the HPI injection valves (which are normally closed). MU-45A need only to maintain the RCS pressure boundary during normal or EOP operation. Removing the valve internals will still allow the valve to maintain the RCS pressure boundary and will not effect the Unit 1 Technical Specifications, Operating License, or the Confirmatory Orders.

☐ Proposed change does not require 10CFR50.59 Evaluation per Attachment 1, Item # ____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in questions 1, 2 and 3. If search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS: 50.59 Unit 1	ALL (MU-45A, MU isolation, HPI w/5 isolation, HPI Nozzle isolation)
MANUAL SECTIONS: 9.1, 4.2.5.2, 6.1.2.1.1	
FIGURES: 4-1	

T. Grant Ehren — Certified Reviewer's Signature / Printed Name — 3/27/2001 Date

Reviewer's certification expiration date: 4/28/2001

Assistance provided by:

Printed Name	Scope of Assistance	Date

Search Scope Review Acceptability (NA, if performed by Technical Reviewer per 1000.006)

Certified Reviewer's Signature — REZA AHRABIE Printed Name — 3/27/2001 Date

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

DM 3/27/01

Document No. ~~ER010269E101~~ DM TAP 01-1-002 Rev./Change No. 0

Complete the following Determination. If the answer to any item below is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 SAFETY EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 3 Pages.

Dwn 3/27/01

TAP01-1-002

Document No. ~~ER010269E101~~ Rev./Change No. 0 10CFR50.59 Eval. No. (Assigned by PSC) FFN # 01-025

Title Evaluate Operation with Valve MU-45A Internals Removed

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

Yes ☐ No ☒

1. Will the probability of an accident previously evaluated in the SAR be increased?

The valve disc is removed from the valve which means the valve will be open all the time and cannot be closed. The safety function on this valve is to maintain pressure boundary which it can do with no disc installed. The valve must be open during power operation since it is on the A HPI injection nozzle line. There will be no loose parts in the valve and the stem will be secure in its position out of the flow stream. Since the valve must be open and without the disc there is no way to close the valve, it will perform its safety function without adverse effects. Therefore, the probability of an accident previously evaluated in the SAR will not be increased.

Yes ☐ No ☒

2. Will the consequences of an accident previously evaluated in the SAR be increased?

This change involves removing the valve disc from a valve that is required to be open during power operation. There is no change in the performance or requirements of the valve. Further MU-45A is not credited with mitigating the consequences of an accident in the SAR. Therefore, the consequences of an accident previously evaluated in the SAR will not be increased.

Yes ☐ No ☒

3. Will the probability of a malfunction of equipment important to safety be increased?

The valve disc is removed form the valve because it was damaged beyond repair, however, the damage to the disc is indeterminate in terms of internal cracking. For this reason, instead of leaving the damaged disc in the valve, it is removed, thus reducing the possibility of introducing loose parts in the RCS. Also the valve is normally open to provide an injection path to the RCS. The valve is capability of performing its safety function, maintain pressure boundary and remain open for safety injection, whether the disc is installed or not. Therefore, the safety function of the valve has not changed and the probability of a malfunction of equipment important to safety will not be increased.

Yes ☐ No ☒

4. Will the consequences of a malfunction of equipment important to safety be increased?

The safety function of this valve will not change. Its safety function is to maintain the RCS pressure boundary and remain open to provide a safety injection path to the RCS. This valve is not credited with mitigating consequences in the event of a safety system malfunction. Therefore, the consequences of a malfunction of equipment important to safety will not be increased.

Yes ☐ No ☒

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created?

MU-45A is designed to maintain the pressure boundary of the RCS and that function will not change. It is also required to remain open to allow an HPI injection flow path to the RCS and this function will also not change. The only effect that removing the valve disc will have on the system is that the valve cannot isolate between the RCS and the HPI injection path. This is not a safety function.

This valve is not credited with any accident mitigation. It is a manually operated valve that is locked open during power operations. The configuration change in the valve, removing the disc, has no impact on plant operations and power operation. There is no change in the valve safety function and no change in the requirements of the valve, therefore, the is no possibility of an accident of a different type than previously evaluated in the SAR to be created.

Yes ☐ No ☒

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created?

Removing the disc from MU-45A will have no impact on the HPI injection path or the RCS system. The stem will be secured by the packing and will not be in the flow path. There are no loose parts in the system created by this change and valve line ups no different than those currently in place. No changes are being made to the RCS or HPI systems that will create a new possible scenario for equipment failure. Therefore, this change does not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR.

Yes ☐ No ☒

7. Will the margin of safety as defined in the basis for any technical specification be reduced?

MU-45A is not credited with any action to mitigate the consequences of an accident or prevent or mitigate the occurrence of a malfunction of equipment important to safety. However, it is does have a passive safety function to maintain the RCS pressure boundary. Removing the disc does not change any of these requirements for the valve. Therefore the only effect the disc removal could have is to cause the valve to be weaker than before so that the margin of safety for pressure boundary requirements is reduced. The valve body will continue to have the same structural integrity as before and not be reduced. There is no reduction of the margin of safety as defined in the basis for any technical specification.

FORM TITLE: 10CFR50.59 SAFETY EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

T. Grant Ehren — Certified Reviewer's Signature

T. Grant Ehren — Printed Name

3/27/2001 — Date

Reviewer's certification expiration date: 4/28/2001

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: _____ Date: 3/27/01

54

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 4 Pages.

Document No. 1628.014 Rev./Change No. 010-00-0

Title Operation of the Oxidizing Biocide System

Brief description of proposed change: Added instructions for contingency treatment of the Unit 1 circulating water system.

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☒ No☐

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. 1628.014 Rev./Change No. 010-00-0

Basis for Determination (Questions 1, 2 & 3):

Description:

Currently, procedure guidance (Attachment 21) is contained in procedure OP-1628.014 (Operation of the Oxidizing Biocide System) for the injection of Biocide into the Circulating Water System using the installed components. The intent of this procedure change is to provide an alternate method of biocide injection if the installed system is not operational.

The contingency method will supply an equivalent biocide injection method using a temporary chemical injection assembly with refillable chemical totes. The injection assembly will be connected to existing Domestic Water hose connections located between the P-3B and P-3C Circulating Water pumps or in the Chemical Addition Pump Room. The injector assembly will use an eductor to inject chemicals (NaBr, NaOCl) from portable tanks into the existing drain holes in the Circulating Water Bays.

Question 1: The Operating Licenses do address biocide addition to the service water but, they do not address biocide addition to the circulating water. Nothing in this procedure change will require a change to the Operating Licenses.
Question 2: The Unit 1 SAR, section 10.4.5 states "The circulating water at the intake structure has a biocide added periodically by the Sodium Bromide/Sodium Hypochlorite System.". This procedure change will allow biocide to be added by a contingency method other than using the Sodium Bromide/Sodium HypoChlorite System. This will require a change to the Unit 1 SAR.
Question 3: This procedure change does not involve a test or an experiment not described in the SAR as defined in Procedure 1000.131.
(See attached 50.59 Evaluation for all yes answers)

☐ Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #_____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS:	
50.59 - Common	**50.59 – Common (biocide*, sodium hypochlorite, sodium bromide, oxidant*, chlorin* or bromin*, NaOCl, NaBr, M-224)**
MANUAL SECTIONS:	
Unit 1 Tech. Spec.	**4.5.2.1.2.a.2**
Unit 2 Tech. Spec.	**4.6.2.3.a.2**
Unit 1 Tech. Spec. Bases	**4.5.2.B**
Unit 2 Tech. Spec. Bases	**3/4.6.2.3.B**
Unit 1 SAR	**9.3.2.1, 10.4.5**
Unit 2 SAR	**9.2.1.2.1, 9.2.1.2.2.1, 9.2.1.3**
FIGURES:	
Unit 1 SAR	**9-10**
Unit 2 SAR	**7.4-2**

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Teresa Madeley — Teresa Madeley — 3/25/01

Certified Reviewer's Signature | Printed Name | Date

Reviewer's certification expiration date: **8/25/01**

Assistance provided by:

Printed Name	Scope of Assistance	Date
Colin Griffin	**Engineering evaluation of installation and operation of contingency equipment.**	**4/13/01**

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

N/A

Certified Reviewer's Signature | Printed Name | Date

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

Document No. **1628.014** Rev./Change No. **010-00-0**

Complete the following Determination. If the answer to any checklist item is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☒	☐	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 2 Pages.

10CFR50.59 Eval. No. FFN #01-028
(Assigned by PSC)

Document No. 1628.014 Rev./Change No. 010-00-0

Title Operation of the Oxidizing Biocide System

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

This procedure change will allow sodium hypochlorite and/or sodium bromide to be added to the circulating water system by a contingency method other than the normal method of adding these chemicals using the Sodium Bromide/Sodium Hypochlorite System. The same chemicals will be added at the same concentrations for the same amount of time to the circulating water system. The installation and operation of the contingency equipment will not adversely affect the systems structures or components evaluated in the SAR. There will be no negative effect on the circulating water system, service water system or the environment. Therefore, the probability of an accident previously evaluated in the SAR will not be increased.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

This procedure change allows chemicals to be added by a different method to the circulating water system. There will be no change in the operation of the circulating water system or the service water system. Therefore, there will be no affect on radiological consequences of an accident previously evaluated in the SAR.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

This procedure change allows sodium hypochlorite and/or sodium bromide to be added to the Unit 1 circulating water by a contingency method. The contingency chemical injection procedure requires continuous monitoring by the chemist and removal of the injection stabs from the CW bays after each injection sequence. The injection stabs are 14 foot long to prevent direct chemical injection on conduits, components and piping in the bay. The length of the injection stabs is greater than four foot above the maximum water level to prevent immersion in turbulent water. The service water system, which contains equipment important to safety, can take suction from the circulating water system. The same chemicals will be added at the same concentrations for the same amount of time. There will be no negative effect on the circulating water system or the service water system. Therefore, the probability of a malfunction of equipment important to safety will not be increased.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

This procedure change will allow sodium hypochlorite and/or sodium bromide to be added to the circulating water system by a contingency method other than the normal method of adding these

chemicals using the Sodium Bromide/Sodium Hypochlorite System. The assembly and operation of this equipment will not negatively impact the consequences of a failure of equipment important to safety. Adding sodium hypochlorite and/or sodium bromide to the circulating water system by a contingency method will have no negative effect on the circulating water or service water systems. Therefore, the dose consequences of a malfunction of equipment important to safety will not be increased.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

The assembly and operation of the contingency equipment will not create the possibility of a different type of accident than previously evaluated in the SAR. Adding sodium hypochlorite and/or sodium bromide to the circulating water system by a different method will have no negative effect on the circulating water system, the service water system, or the environment. Therefore, the possibility of an accident of a different type than any previously evaluated in the SAR will not be created.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

The assembly and operation of the contingency equipment will not increase the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the SAR. The same chemicals will be added at the same concentration for the same amount of time to the circulating water system. The only equipment important to safety that could potentially be involved is equipment in the service water system since the service water system can draw suction from the circulating water bays. The circulating water system is not a system important to safety. There will be no negative effect on the operation of the circulating water system or the service water system. Malfunction of service water equipment has been previously evaluated in the SAR. Therefore, the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR will not be created.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

This procedure change allows the installation and operation of the contingency method of biocide injection equipment. The assembly and operation of the contingency equipment will not reduce the margin of safety and will have no negative effect on the operation of the circulating water system or the service water system. Therefore, no margin of safety will be reduced.

Teresa Madeley — Certified Reviewer's Signature

Teresa Madeley — Printed Name

3/25/01 — Date

Reviewer's certification expiration date: 8/25/01

Assistance provided by:

Printed Name	Scope of Assistance	Date
Colin Griffin	Engineering evaluation of installation and operation of contingency equipment.	4/13/01

PSC review by: _____ Date: 4/24/01

55

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 4 Pages.

Document No. DRN 01-644 Rev./Change No. 0

Title **Change Description and Valve Lineup Position for IA-611**

Brief description of proposed change: **IA-611 currently serves as an isolation valve for a short capped Instrument Air line, however, it is depicted on P&ID M-218 sheet 4 and SAR figure 9-14 sheet 4 as feeding valves in the Unit One Hot Lab. The valves were removed and the line capped many years ago. This DRN justifies the position of IA-611 as a normally closed valve on the SAR drawing, P&ID, and in OPS procedure 1104.024 valve lineup.**

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. **DRN 01-644** Rev./Change No. **0**

Basis for Determination (Questions 1, 2 & 3):

☐ Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #_____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS:	
Unit 1 50.59 Documents	**All (IA-611), (Instrument Air)**

MANUAL SECTIONS:
9.9

FIGURES:
9-14 sheet 4

Bradley W Short — Bradley W Short — 5-15-01
Certified Reviewer's Signature — Printed Name — Date

Reviewer's certification expiration date: 9/20/01

Assistance provided by:

Printed Name	Scope of Assistance	Date
Tom Van Schaik	**Reference searches**	**5/1/01**

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

[signature] — David N. McKearney — 5/16/01
Certified Reviewer's Signature — Printed Name — Date

ENVIRONMENTAL IMPACT DETERMINATION
(UNIT 1 and UNIT 2)

Document No. **DRN 01-643** Rev./Change No. **0**

Complete the following Determination. If the answer to any checklist item is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils; surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

FORM TITLE:	FORM NO.	REV.
10CFR50.59 SAFETY EVALUATION	1000.131B	003-04-0

This Document contains 2 Pages.

Document No. DRN 01-644 Rev./Change No. 0 10CFR50.59 Eval. No. FFN#01-031 (Assigned by PSC)

Title Change Description and Valve Lineup Position for IA-611

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   The position of IA-611 is unrelated to any of the accidents evaluated in the SAR. Maintaining the isolation valve to a short capped line shut will not increase the probability of any accident evaluated in the SAR.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   The position of IA-611 has no impact on gaseous or liquid radwaste system inventory nor does it affect the dose consequences of any accident analyzed in the SAR. Maintaining IA-611 in the closed position is thus immaterial.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   The Instrument Air system itself is non safety significant, so by having a capped line additionally isolated and eliminating that source of air leakage as well, there can be no increase in probability of malfunction of equipment important to safety.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   If equipment important to safety did malfunction, there is no logical tie to having this isolation valve shut that would lead to that malfunction and increase radiological release consequences.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   Changing the position of IA-611 cannot initiate an accident of a different type — loss of Instrument Air is common to several DBAs but not as an initiator. Changing this valve to being normally closed would, in any case, be a conservative action.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   No adverse effects on equipment important to safety are possible due to maintaining IA-611 normally closed, as opposed to open. Therefore, the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the SAR will not be created due to this change.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

This change has no effect on a margin of safety as defined in the basis for any technical specification. Therefore, this change will not result in a reduction to a margin of safety as defined in the basis for any technical specification.

Bradley W Short — Certified Reviewer's Signature
Bradley W Short — Printed Name
5-15-01 — Date

Reviewer's certification expiration date: 9/20/01

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: _____ Date: 5/31/01

56

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 4 Pages.

Document No. -ER #002545E101 Rev./Change No. 0

Title Incorporate SQUG/GIP/USI A-46 Seismic Qualification Methods into the ANO-1 SAR

Brief description of proposed change: This ER Evaluation/50.59 demonstrates that it is acceptable to use earthquake and seismic testing experience as an alternative method for seismic design and verification of new, modified and replacement equipment (e.g., seismic equipment qualification) at ANO-1. This methodology was approved and endorsed by the NRC for use in resolving Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Electrical and Mechanical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46". The NRC approved this methodology in their SSER No.2 to the Generic Implementation Procedures (GIP). ANO-1 has successfully completed the requirements of the GL and has received its Safety Evaluation Report (SER) from the NRC (Licensing Letter #0CNA020003, dated 2/7/00). In the SER, the NRC notes that ANO-1 may revise its licensing basis in accordance with 10 CFR 50.59 to incorporate the GIP methodology. This ER/50.59 Evaluation implements the change to the ANO-1 licensing basis (i.e., SAR). NOTE: This ER/50.59 Evaluation makes NO PHYSICAL CHANGES to the plant. It DOES NOT replace or supercede EXISTING licensing basis methods for the seismic qualification of equipment at ANO-1 (e.g. IEEE 344 1971). It recognizes that earthquake and seismic testing experience is an ADDITIONAL/ALTERNATIVE method of seismic qualification of equipment at ANO-1.

Will the proposed Activity:

1. Require a change to the Operating License including:

Technical Specifications (excluding the bases)?	Yes☐ No☒
Operating License?	Yes☐ No☒
Confirmatory Orders?	Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

SAR (multi-volume set for each unit)?	Yes☒ No☐
Core Operating Limits Report	Yes☐ No☒
Fire Hazards Analysis?	Yes☐ No☒
Bases of the Technical Specifications?	Yes☐ No☒
Technical Requirements Manual?	Yes☐ No☒
NRC Safety Evaluation Reports?	Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. ER #002545E101 Rev./Change No. 0

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:

   QAMO? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. ER #002545E101 Rev./Change No. 0

Basis for Determination (Questions 1, 2 & 3):
ZYFIND Search on LRS system was performed as indicated in the Search Scope section. The ANO-1 SAR is being changed to permit the use of earthquake and seismic testing experience as an acceptable and alternative method of seismic qualification of equipment at ANO-1. The change only involves the ANO-1 SAR. Neither the ANO-1 Tech Specs, the Operating License, nor any Confirmatory Orders are impacted by this change because they do not address seismic design basis issues. In addition, and with the exception of the ANO-1 SAR, none of the other documents listed in question 2 are impacted by the change, because they too do not address seismic design basis issues. Lastly, this change does not involve a test or experiment not described in the SAR, nor does it impact the environment since no physical changes to the plant/plant site are involved.

☐ Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #_____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS: Keyword Search Strings: Scope of Search = 50.59 ANO-1; Key words searched include "Seismic", "Seismic Qualification", "IEEE344-1971", "IEEE344-1975", "Maximum Hypothetical Earthquake", "equipment w/10 qualification", "seismically", "seismic w/10 qualification", "seismically w/10 qualified" and "Seismic Category 1".	

MANUAL SECTIONS: Tech Specs/Tech Spec Bases, Operating License and COLR. The Technical Requierements Manual and SAR Sections 2.6, 7.1.1, 8.1.4, A.5 & Chapter 5.0

FIGURES: SAR Figures 5-10, 5-11, 5-12, 5-13 & 5-14

Certified Reviewer's Signature	David J. Lach Printed Name	1/31/01 Date

Reviewer's certification expiration date: 2/05/2003

Assistance provided by:

Printed Name	Scope of Assistance	Date

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

Certified Reviewer's Signature	Doyle G. Adams Printed Name	2/22/01 Date

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

ENVIRONMENTAL IMPACT DETERMINATION
(UNIT 1 and UNIT 2)

Document No. ER #002545E101 Rev./Change No. 0

Complete the following Determination. If the answer to any checklist item is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 3 Pages.

10CFR50.59 Eval. No. FFN# 01-034
(Assigned by PSC)

Document No. ER #002545E101 Rev./Change No. 0

Title Incorporate SQUG/USI A-46 Seismic Qualification Methods into the ANO-1 SAR

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

The response to this question addresses the impact of the proposed change on a seismic event both as a potential accident initiator, and as an occurrence considered in equipment design. The proposed change involves allowance of the GIP method as an alternative method for demonstration of seismic adequacy of equipment.

The only accidents in the SAR that could potentially be affected by the use of the GIP method are the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE) (a.k.a. "Design Earthquake" & "Maximum Earthquake", respectively, in the ANO-1 SAR). Earthquakes are considered to be acts of nature (or natural phenomena) and are not controllable. Consequently, the use of earthquake and seismic testing experience as a method of seismic equipment qualification cannot have any bearing on the probability of an earthquake occurring. Therefore, the use of this methodology does not, in any manner, increase the probability of occurrence of either of the ANO-1 design basis earthquakes.

Relative to the current ANO-1 licensing basis, it is demonstrated that the GIP method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and after a seismic event. As such, the proposed change has no impact on a seismic event as an occurrence considered in equipment design. The use of the GIP methodology specifically considers and includes the seismic event as a design basis occurrence.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

The proposed change involves allowance of the GIP method as an alternative method for demonstration of seismic adequacy of equipment. Relative to the current ANO-1 licensing basis, it has been demonstrated that the GIP method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and after a seismic event. As such, assumptions in previously analyzed accidents in the SAR regarding availability and performance of equipment to mitigate an accident following a seismic event are unchanged. Therefore, the proposed change does not increase the consequences of an accident previously evaluated in the SAR.

The only accidents in the SAR that could potentially have radiological release consequences affected by the use of the GIP method are those accidents analyzed in the SAR associated with the Operating Basis Earthquake and the Safe Shutdown Earthquake. The use of a new method for demonstrating equipment seismic adequacy could *potentially* affect the ability of safety-related equipment or equipment important to safety to perform required safety functions during or after a seismic event, thus affecting radiological release consequences. However, because the use of the GIP methodology provides equivalent or superior

assurance of equipment seismic adequacy to that provided by the current ANO-1licensing basis, the proposed change will have no effect on and will change no accident consequences. For that same reason, the use of earthquake and seismic testing experience for seismic equipment qualification will have no effect on radiological release consequences.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

The proposed change recognizes the GIP method as an alternative method for demonstration of seismic adequacy of equipment. Relative to the current ANO-1 licensing basis, it has been demonstrated that the GIP method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and after a seismic event. As such, there is no decrease in the seismic adequacy of equipment.

Because the GIP method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and after a seismic event, no equipment important to safety is affected by the proposed change. In addition, as noted above, because there is no decrease in the seismic adequacy of equipment, any such equipment item will continue to perform required safety functions during and after the earthquake. The result is no increase in the probability of a malfunction of equipment important to safety as a result of a seismic event. Therefore, the proposed change will not increase the probability of occurrence of a malfunction of equipment important to safety.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

The proposed change involves allowance of the GIP method as an alternative method for demonstration of seismic adequacy of equipment. Relative to the current ANO-1 licensing basis, the GIP method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and after a seismic event. As such, there is no decrease in the seismic adequacy of equipment.

Therefore, since there are no adverse effects on the ~~the~~ seismic adequacy of equipment as a result of this change, the proposed change will not increase the consequences of a malfunction of equipment important to safety.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

The proposed change recognizes the GIP method as an alternative method for demonstration of seismic adequacy of equipment. Relative to the current ANO-1 licensing basis, the GIP method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and after a seismic event.

The ANO-1 SAR requirements regarding seismic adequacy of equipment include definition of the subset of equipment which must meet seismic adequacy requirements (via the Q-List) and definition of the method for demonstrating seismic adequacy (Section 5.1.4.2 of the SAR). The proposed change provides an alternative method for demonstrating seismic adequacy and does not change the subset of equipment which must meet seismic adequacy requirements. Since the GIP method provides an equivalent or superior level of assurance of seismic adequacy relative to the current licensing basis, the proposed change will continue to assure regulatory requirements regarding seismic adequacy of equipment are met.

Since the proposed change does not affect the set of equipment which must meet seismic adequacy requirements or the level of seismic adequacy as defined in the SAR, the proposed change does not create the possibility of an accident of a different type than previously evaluated in the SAR.

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

Document No. ER #002545E101 Rev./Change No. 0

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

The proposed change involves allowance of the GIP method as an alternative method for demonstration of seismic adequacy of equipment. Relative to the current ANO-1 licensing basis, the GIP method provides an equivalent or superior level of assurance that equipment will perform required safety functions during and after a seismic event.

The GIP method addresses specific seismic failure modes identified during real earthquakes, that are not specifically addressed in the current ANO-1 licensing basis method. However, in identifying the potential seismic failure modes, the GIP method also provides guidelines, caveats and criteria that provide equivalent or superior levels of assurance that the equipment will withstand the various potential seismic failure modes. Consideration of these specific seismic failure modes does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the SAR. Rather, it actually reduces the possibility of equipment malfunctions resulting from seismic events because the GIP method provides the guidelines to prevent the malfunction (due to identified seismic failure modes) from ever occurring in the first place. Therefore, the proposed change will not introduce any new equipment failure modes and thus does not create the possibility of a malfunction of equipment important to safety of a different type than previously evaluated in the SAR.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

The proposed change recognizes the GIP method as an alternative method for demonstration of seismic adequacy of equipment. Relative to the current ANO-1 licensing basis, the GIP method results in an equivalent or superior level of assurance that equipment will perform required safety functions during and after a seismic event. As such, there is no decrease in the seismic adequacy of equipment. With no reduction in the ability of equipment to withstand a seismic event, there is no reduction in the margin of safety for the equipment item. This is true literally; (i.e., its seismic design margin is not impacted and therefore there are no impacts to the physical parameters of equipment that define its performance of safety limits or protective boundaries during a seismic event). This is also true for any upper level design margins as defined in the bases for any Tech Spec (i.e., any equipment item, specified in the Tech Spec bases to safely shut the plant down, or relied upon in the Tech Spec bases to perform required safety functions, will remain fully functional during and after the seismic event).

Furthermore, to demonstrate that the GIP method does not result in a reduction of safety margin relative to the ANO-1 licensing basis, a comparison between the GIP method and the ANO-1 licensing basis was made. This comparison is documented in Table 1 of the ER Evaluation. Differences between the GIP method and the ANO-1 licensing basis were identified and the effect of the differences on the overall cumulative relative safety margin was determined. The results demonstrate that the use of the GIP method will not reduce the plant margin of safety

Certified Reviewer's Signature	David J. Lach Printed Name	1/31/01 Date

Reviewer's certification expiration date: 2/05/2003

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: _____ Date: 6/14/01

57

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 3 Pages.

Document No. COOP Log (cyclone separators) Rev./Change No. 0

Title Circ Water Pump Cyclone Separator Bypass

Brief description of proposed change: The cyclone separators for the circ water pumps bearing lubrication system have been bypassed for greater than one year. This is an evaluation of this condition since the cyclone separators are shown in service on SAR figure 9-10. The valves out of position with respect to SAR figure 9-10 are SW-96A, SW-96B, SW-96C, SW-4A, SW-4B, and SW-4C. The service water system normally supplies the water for the circulating water pump bearing lubrication system through a flow limiting orifice as shown on P&ID M-209, sheet 1.

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. **COOP Log (cyclone separators)** Rev./Change No. **0**

Basis for Determination (Questions 1, 2 & 3):

1. The circulating pump bearing lubrication supply is not addressed in the operating license.

2. SAR section 9.3.2.1 and ANO-1 Safety Evaluation Report section 9.3.1 both state that service water supplies cooling to the circulating water pump bearing lubrication system. This change does not make this untrue. SAR figure 9-10 shows cyclone separators in service in this cooling water supply. This change removes the separators from service by closing the inlet valves and opening the bypasses.

3. The change involves normal plant operation only and no test or experiment.

4. The change results in no impact to the environment. See the environmental impact determination.

5. The change does not involve radioactive material.

6. The change has no effect on ventilated storage cask activities.

7. There is not mention of the circulating water pump bearing lubrication system in the QAPM or the E-plan.

8. The change is not dependent on pending NRC approvals of any other action.

☐ Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #_____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS:	
All	**LRS Unit 1 50.59 search index (CW, circulat*, circ)**

MANUAL SECTIONS:
SAR figures

FIGURES:
SAR Figure 9-10

Certified Reviewer's Signature	Printed Name	Date
Alan Cox	**Alan Cox**	**5/28/01**

Reviewer's certification expiration date: **01/05/02**

Assistance provided by:

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 3 PC-1,2

Printed Name	Scope of Assistance	Date
N/A		

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

William R. Rowlett, Jr. — Certified Reviewer's Signature

William R. Rowlett, Jr. — Printed Name

5/31/2001 — Date

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.69 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

ENVIRONMENTAL IMPACT DETERMINATION
(UNIT 1 and UNIT 2)

Document No. COOP log (cyclone separators) Rev./Change No. 0

Complete the following Determination. If the answer to any checklist item is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 2 Pages.

10CFR50.59 Eval. No. FFN #01-036
(Assigned by PSC)

Document No. **COOP Log (cyclone separators)** Rev./Change No. **0**

Title **Circ Water Pump Cyclone Separator Bypass**

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   Operation of the circulating water system or failure of the system is not involved in the initiation of an accident evaluated in the SAR.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   Operation of the circulating water system is not credited with mitigating the consequences of an accident evaluated in the SAR. Bypassing the cyclone separators has an insignificant impact on flow rate from the service water system.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   The circulating water pumps and the cyclone separators are not considered equipment important to safety. In addition, no increase in failure probability of the circulating water pumps has been noted due to bypassing the separators.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   The circulating water pumps are non-safety related and no equipment important to safety is postulated to be impacted by bypassing the separators.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   Bypassing the separators is expected to effect no equipment other than the circulating water pumps. The malfunction or failure of these pumps cannot create the possibility of an accident of a different type than previously evaluated in the SAR.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

Bypassing the cyclone separators could lead to the possibility of malfunction of the circulating water pumps. However, this malfunction would not be a different type than malfunctions of many other non-safety related components.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

No margin of safety is defined in the bases of the Technical Specifications related to the circulating water pumps or the cyclone separators associated with those pumps.

Certified Reviewer's Signature	Printed Name	Date
	Alan Cox	**02/28/01**

Reviewer's certification expiration date: **01/05/02**

Assistance provided by:

Printed Name	Scope of Assistance	Date
N/A		

PSC review by: _____ Date: 6-21-01

58

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 3 Pages.

Document No. COOP Log (TB drain rad mon) Rev./Change No. **0**

Title **Turbine building drain radiation monitor out of service**

Brief description of proposed change: **The turbine building drain radiation monitor has been removed from service by closure of valves SS-885 and SS-889. In addition, the handswitch, HS-5641, is in the off position. These components remove RE-5641 from service contrary to SAR table 11-7.**

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete the Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7:

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. **COOP Log (TB drain rad mon)** Rev./Change No. **0**

Basis for Determination (Questions 1, 2 & 3):

1. The turbine building drain radiation monitor is not addressed in the operating license.

2. SAR Table 11-7 lists turbine building drain line process monitor. This change removes the monitor from service.

3. The change involves normal plant operation only and no test or experiment.

4. The change results in no impact to the environment. See the environmental impact determination.

5. The change does not involve processing radioactive material outside of normal areas.

6. The change has no effect on ventilated storage cask activities.

7. There is not mention of the turbine building drain radiation monitor system in the QAPM or the E-plan.

8. The change is not dependent on pending NRC approvals of any other action.

☐ Proposed change does not require 10 CFR 50.59 Evaluation per Attachment 1, Item #_____, (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in Question 1, 2 and 3. If a search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS: **All**	**LRS Unit 1 50.59 search index (turbine building drain, radiation monitor*, RE-5641)**
MANUAL SECTIONS: **SAR**	**Table 11-7**
FIGURES: **SAR figures**	

Certified Reviewer's Signature | **Alan Cox** Printed Name | **5/29/01** Date

Reviewer's certification expiration date: **01/05/02**

Assistance provided by:

Printed Name	Scope of Assistance	Date
N/A		

Search Scope Review Acceptability (NA, if performed by Technical Review per 1000.006)

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

William R. Rowlett, Jr. — Certified Reviewer's Signature

William R. Rowlett, Jr. — Printed Name

5/31/2001 — Date

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 2 Pages.

10CFR50.59 Eval. No. FFN#01-037
(Assigned by PSC)

Document No. **COOP Log (TB drain rad mon)** Rev./Change No. **0**

Title **Turbine building drain radiation monitor out of service**

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   The turbine building drain radiation monitor is a monitoring system only and therefore is not involved in the initiation of an accident evaluated in the SAR.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   Operation of the turbine building drain radiation monitor is not credited with mitigating the consequences of an accident evaluated in the SAR. Therefore, removing the monitor from service will not increase the consequences of an accident.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   The turbine building drain radiation monitor is not considered equipment important to safety. Removing the monitor from service will have no effect on equipment important to safety.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   The turbine building drain radiation monitor is non-safety related and no equipment important to safety is postulated to be impacted by removing it from service.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   Removing the turbine building drain radiation monitor from service is expected to effect no other equipment. The absence of the turbine building drain radiation monitor will not create the possibility of an accident of a different type than previously evaluated in the SAR.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

Removing the turbine building drain radiation monitor from service will not create the possibility of a malfunction important to safety. The turbine building drain radiation monitor is located in the turbine building far removed from equipment important to safety.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

No margin of safety is defined in the bases of the Technical Specifications related to the turbine building drain radiation monitor.

Certified Reviewer's Signature	Alan Cox Printed Name	02/29/01 Date

Reviewer's certification expiration date: **01/05/02**

Assistance provided by:

Printed Name	Scope of Assistance	Date
N/A		

PSC review by: _____ Date: 6-21-01

59

ARKANSAS NUCLEAR ONE		Page 1
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

This Document contains 3 Pages.

Document No. ER974259N101 Rev./Change No. 0

Title Evaluate Components in the Scope of Penetration P 41

Brief description of proposed change:

Credit is no longer being taken for CV-1667 to close on an ES signal to provide containment isolation for penetration 41. N2-47, a locked closed manual valve, now performs the containment isolation function for outside containment for P- 41. The LBDs need to be updated to reflect relocation of the containment isolation boundary from CV-1667, to N2-47.

Will the proposed Activity:

1. Require a change to the Operating License including:

   Technical Specifications (excluding the bases)? Yes☐ No☒

   Operating License? Yes☐ No☒

   Confirmatory Orders? Yes☐ No☒

2. Result in information in the following SAR documents (including drawings and text) being (a) no longer true or accurate, or (b) violate a requirement stated in the document:

   SAR (multi-volume set for each unit)? Yes☒ No☐

   Core Operating Limits Report? Yes☐ No☒

   Fire Hazards Analysis? Yes☐ No☒

   Bases of the Technical Specifications? Yes☐ No☒

   Technical Requirements Manual? Yes☐ No☒

   NRC Safety Evaluation Reports? Yes☐ No☒

3. Involve a test or experiment not described in the SAR? (See Attachment 2 for guidance) Yes☐ No☒

4. Result in a potential impact to the environment? (Complete Environmental Impact Determination of this form.) Yes☐ No☒

5. Result in the need for a Radiological Safety Evaluation per section 6.1.5? Yes☐ No☒

6. Result in any potential impact to the equipment or facilities utilized for Ventilated Storage Cask activities per Section 6.1.6? Yes☐ No☒

7. Involve a change under 10CFR50.54 for the following SAR documents per Section 6.1.7?

   QAPM? Yes☐ No☒

   E-Plan? Yes☐ No☒

8. Does this review depend on future NRC approval of other actions? (NRC SER, Relief, etc)? (forward change to PSC per 6.3.8 or 6.3.9) Yes☐ No☒

ARKANSAS NUCLEAR ONE		Page 2
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

Document No. ER974259N101 Rev./Change No. 0

Basis for Determination (Questions 1, 2, & 3):

Question 1: The Operating License does not address the components contained in penetration 41, nor does it provide details pertaining to CV-1667 or N2-47. Moving the containment isolation boundary from CV-1667 to N2-47 will not make the Operating License untrue.

Question 2: SAR Table 5-1, Table 7-5, Table 9-25, Figure 9-4, and Section 5.2.2.4.1 will require revision.

Question 3: The proposed change does not involve a test or experiment. The nitrogen supply to the reactor building will not be operated in modes that have not been previously analyzed.

☐ Proposed change does not require 10CFR50.59 Evaluation per Attachment 1, Item #____. (If checked, note appropriate item #, send LDCR to Licensing).

Search Scope:

List sections reviewed in the Licensing Basis Documents specified in questions 1, 2 and 3. If search was performed on LRS, the LRS search index should be entered under "Section" with the search statement(s) used in parentheses. Controlled hard copies of the documents shall be reviewed (LRS is not verified and searches only text, not figures or drawings). **Attach and distribute a completed LDCR per Section 6.1.2 if LBD changes are required.**

Document	Section
LRS: ZY Index Unit 1 50.59	ALL (nitrogen, CV-1667, CV1667, containment w/10 penetration, containment w/10 isolation, diverse w/10 isolation, ESF w/10 isolation)
MANUAL SECTIONS:	SAR 1.4.47, 1.4.49, 5.2.2, 5.2.2.4.1, 5.2.5, Table 5-1, 7.1.3.2.2, 7.1.3.2.3, Table 7-5, Table 7-11A, 9.10, Table 9-25
FIGURES:	Figure 9-4

/ Danielle Smith — Danielle J. Smith — 7/31/2001
Certified Reviewer's Signature / Printed Name / Date

Reviewer's certification expiration date: 11/17/2002

Assistance provided by:

Printed Name	Scope of Assistance	Date

Search Scope Review Acceptability (NA, if performed by Technical Reviewer per 1000.006)

Saif Khan — Saif Khan — 7/31/2001
Certified Reviewer's Signature / Printed Name / Date

ARKANSAS NUCLEAR ONE		Page 3
FORM TITLE: 10CFR50.59 DETERMINATION	FORM NO. 1000.131A	REV. 003-04-0

ENVIRONMENTAL IMPACT DETERMINATION (UNIT 1 and UNIT 2)

Document No. ER974259N101 Rev./Change No. 0

Complete the following Determination. If the answer to any item below is "Yes", an Environmental Evaluation is required. See Section 6.1.4 for additional guidance.

Will the Activity being evaluated:

Yes	No	
☐	☒	Disturb land that is beyond that initially disturbed during construction (i.e., new construction of buildings, creation or removal of ponds, or other terrestrial impact)? See Unit 2 SAR Figure 2.5-17. This applies only to areas outside the protected area.
☐	☒	Increase thermal discharges to lake or atmosphere?
☐	☒	Increase concentration of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Increase quantity of chemicals to cooling lake or atmosphere through discharge canal or tower?
☐	☒	Modify the design or operation of cooling tower which will change drift characteristics?
☐	☒	Install any new transmission lines leading offsite?
☐	☒	Change the design or operation of the intake or discharge structures?
☐	☒	Discharges any chemicals new or different from that previously discharged?
☐	☒	Potentially cause a spill or unevaluated discharge which may effect neighboring soils, surface water or ground water?
☐	☒	Involve burying or placement of any solid wastes in the site area which may effect runoff, surface water or ground water?
☐	☒	Involve incineration or disposal of any potentially hazardous materials on the ANO site?
☐	☒	Result in a change to nonradiological effluents or licensed reactor power level?
☐	☒	Potentially change the type or increase the amount of non-radiological air emissions from the ANO site.

ARKANSAS NUCLEAR ONE		
FORM TITLE: 10CFR50.59 SAFETY EVALUATION	FORM NO. 1000.131B	REV. 003-04-0

This Document contains 2 Pages.

Document No. ER974259N101 Rev./Change No. 0 10CFR50.59 Eval. No. FFN# 01-038 (Assigned by PSC)

Title Evaluate Components in the Scope of Penetration P 41

A WRITTEN RESPONSE PROVIDING THE BASIS FOR THE ANSWER TO EACH QUESTION MUST BE ATTACHED. EACH QUESTION MUST BE ANSWERED SEPARATELY. A SIMPLE STATEMENT OF CONCLUSION IS NOT SUFFICIENT. ATTACHMENT 2 PROVIDES GUIDANCE FOR RESPONSE.

If the answer to any question on this form is "Yes," then an unreviewed safety question is involved. If the answer to all questions is "No," then the proposed change does not involve an unreviewed safety question.

1. Will the probability of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   The proposed change involves components in the nitrogen supply to the reactor building. This system is non-Q and is not credited with initiating any of the accidents evaluated in the SAR. Relocating the isolation boundary for P-41 will not create any new conditions that would increase the likelihood of the events which are credited with initiating an accident previously evaluated in the SAR.

2. Will the consequences of an accident previously evaluated in the SAR be increased? Yes ☐ No ☒

   N2-47 is a locked closed containment isolation valve, which is used infrequently when containment integrity is required. Additionally, it is administratively controlled to ensure that the valve is closed in a timely manner in the event that the valve is in use at the time of an accident. Therefore, there is no increase in the offsite dose consequences of a previously analyzed accident as a result of relocating the isolation boundary from CV-1667 to N2-47.

3. Will the probability of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   Moving the isolation boundary to N2-47 does not degrade the performance of P-41. Since N2-47 is a locked closed manual valve and is located between CV-1667 and P-41, it meets the requirements of GDC-56, which states that the outside containment isolation valve can be a locked closed isolation valve or an automatic isolation valve and shall be located as close to containment as practical. In addition, N2-47 is a 1" 1500 lb ASME Class 1 valve which exceeds the line class requirements and it is periodically tested per Appendix J requirements.

4. Will the consequences of a malfunction of equipment important to safety be increased? Yes ☐ No ☒

   N2-47 (isolation outside containment) and N2-32 (isolation inside containment) are included in the Appendix J and IST programs to verify that they meet the leak rate criteria. There will be no change in the offsite radiation dose (i.e., consequences of a failure) associated with a plant's response to an accident as result of moving the containment isolation boundary to N2-47 since the leak rate acceptance criteria has not changed.

5. Will the possibility of an accident of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   Moving the isolation boundary will not create an accident of a different type since no new failures are introduced due to this change. N2-47 is a locked closed isolation valve which meets GDC 56 requirements and is therefore an acceptable boundary isolation valve.

6. Will the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created? Yes ☐ No ☒

   Moving the isolation boundary to N2-47 does not introduce a malfunction that has not been previously evaluated. N2-47 is located in a seismically qualified section of piping, and is included in the Appendix J program.

7. Will the margin of safety as defined in the basis for any technical specification be reduced? Yes ☐ No ☒

   The Tech Spec basis for Reactor Building Integrity (Section 3.6) states that when reactor building integrity is established, the limits of 10CFR100 will not be exceeded should a maximum hypothetical accident occur. Since integrity of P-41 is verified during the performance of the LLRT and since the acceptance criteria has not changed for this penetration, the margin of safety has not been reduced by moving the isolation boundary to N2-47.

Danielle Smith — Certified Reviewer's Signature

Danielle J. Smith — Printed Name

7/31/2001 — Date

Reviewer's certification expiration date: 11/17/2002

Assistance provided by:

Printed Name	Scope of Assistance	Date

PSC review by: RV Fuller Date: 8-2-01